NUCLEAR SAFETY

OF MICHIGAN
FEB 22 1962
PHOENIX

Phoenix

October 1961

TECHNICAL PROGRESS REVIEWS

To meet the needs of industry for concise summaries of current atomic developments, the Atomic Energy Commission is publishing this series, Technical Progress Reviews. Issued quarterly, each of the reviews digests and evaluates the latest findings in a specific area of nuclear technology and science.

The four journals published in this series are:

Nuclear Safety, W. B. Cottrell and associates, Oak Ridge National Laboratory

Power Reactor Technology, Walter H. Zinn and associates, General Nuclear Engineering Corporation

Reactor Core Materials (covering solid material developments), R. W. Dayton, E. M. Simons, and associates, Battelle Memorial Institute

Reactor Fuel Processing, Stephen Lawroski and associates, Chemical Engineering Division, Argonne National Laboratory

Each journal may be purchased ($2.00 per year for subscription and individual issues $0.55) from the Superintendent of Documents, U. S. Government Printing Office, Washington 25, D. C. See back cover for remittance instructions and foreign postage requirements.

The views expressed in this publication do not necessarily represent those of the United States Atomic Energy Commission, its divisions or offices, or of any Commission advisory committee or contractor.

Availability of Reports Cited in This Review

Unclassified AEC reports are available for inspection at AEC depository libraries and are sold by the Office of Technical Services, Department of Commerce, Washington 25, D. C. Some of the reports cited are not available owing to their preliminary nature; however, the information contained in them will eventually be made available in formal progress or topical reports.

Unclassified reports issued by other Government agencies or private organizations should be requested from the originator.

Unclassified British and Canadian reports may be inspected at AEC depository libraries. British reports are sold by the British Information Service, 45 Rockefeller Plaza, New York, N. Y.; Canadian reports (AECL series) are sold by the Scientific Document Distribution Office, Atomic Energy of Canada, Ltd., Chalk River, Ontario, Canada.

Classified U. S. and foreign reports identified in this journal as Classified may be purchased by properly cleared Access Permit Holders from the Division of Technical Information Extension, U. S. Atomic Energy Commission, P. O. Box 1001, Oak Ridge, Tenn. Such reports may be inspected at classified AEC depository libraries.

NUCLEAR SAFETY

A REVIEW OF RECENT DEVELOPMENTS

Prepared for DIVISION of TECHNICAL INFORMATION,
U. S. ATOMIC ENERGY COMMISSION,
by OAK RIDGE NATIONAL LABORATORY

● DEC. 1961

● VOLUME 3

● NUMBER 2

foreword

This quarterly journal is a Technical Progress Review prepared by the Oak Ridge National Laboratory at the request of the Division of Technical Information, U. S. Atomic Energy Commission. This Review is intended to assist those interested in keeping abreast of significant developments in the field of nuclear safety. *Nuclear Safety* is not a comprehensive abstract of all literature published in this field during a given quarter; rather it is a mechanism for presenting concise reviews of selected subjects as prevailing interest and available information warrant.

Coverage of the Review is limited to topics relevant to the analysis and control of hazards associated with nuclear reactors, operations involving fissionable materials, and the products of nuclear fission. Primary emphasis is on safety in reactor design, construction, and operation; however, safety considerations in reactor fuel fabrication, spent-fuel processing, nuclear waste disposal, and related operations are also treated. Safety in the use of radioisotopes in industry, medicine, and research is excluded, as are most topics considered to be the province of health physics. Even with these exclusions, nuclear safety overlaps such diverse fields as nuclear physics, solid-state physics, mechanics, chemistry, meteorology, geology, seismology, metallurgy, law, and nearly all branches of engineering. The authors will therefore review material from these fields which, in their opinion, has a direct bearing on nuclear safety.

Two distinctly different types of articles are in this issue of *Nuclear Safety:* reviews of current literature and special review articles on specific topics. The editors feel that each of these articles makes a necessary and distinctive contribution to this journal. The special review articles permit discussion of pertinent subjects which cannot be adequately considered by reference to only the current literature. The current review articles, however, constitute the major portion of this issue. All incoming literature (including reports, books, American and foreign technical journals, and transactions) is examined for subjects within our area of interest. This material is collected, grouped, and reviewed by experts. Interpretations, in any article, represent the opinions of the editors, who are employees of the Oak Ridge National Laboratory. Readers are urged to consult the references to original work for more complete information.

It is recognized that the critical evaluation of subject areas leading to the determination of criteria cannot fail to stimulate contrary opinions. This is expected to be particularly true in the area of nuclear safety since, in many instances, only preliminary information is available, the ramifications are many and varied, and opinion and judgment must be relied upon so heavily. Although the editors do not propose that *Nuclear Safety* act as a clearing house for safety correspondence because of the above facts, we have had for some time a policy that would permit the publication of statements of position at variance with those expressed by the editors. Such statements will be published after the editors have ascertained that a real difference exists and that the position is reasonable.

In addition to the invited contributors, many members of the Oak Ridge National Laboratory staff wrote review material, reviewed manuscripts, or otherwise contributed to this publication. Their contributions are gratefully acknowledged.

W. B. COTTRELL, Editor
C. G. BELL, E. E. GROSS, W. de LAGUNA,
A. W. SAVOLAINEN, and C. S. WALKER, Assistant Editors
Oak Ridge National Laboratory

DEPOSITED BY THE
UNITED STATES OF AMERICA

contents

Foreword

1 I GENERAL SAFETY CRITERIA

1 Reactors, Sites, and Safety
4 Nuclear Space Systems
6 JCAE Hearings on Radiation Safety and Regulation
6 Breeder Reactors and Nuclear Safety
8 Safety in Source Handling
13 References

15 II ACCIDENT ANALYSIS

15 Fission-Product Release from UO_2
20 Activity Buildup in Pressurized-Water Reactor Systems
22 Studies in Reactor Kinetics
29 References

32 III CONTROL AND INSTRUMENTATION

32 The Karlsruhe Symposium on Criticality Control
39 Measurement of Control-Rod Effectiveness
41 In-Core Flux Monitoring
42 Nonnuclear In-Core Instrumentation
44 References

46 IV PLANT SAFETY FEATURES

46 Deposition of Radioactive Materials from Gases
49 Effects of Blasts on Containment Vessels
51 Beryllium Hazards in Nuclear Operations
54 References

56 V CONSEQUENCES OF ACTIVITY RELEASE

56 Chemical and Radiation Induced Genetic Exposures
58 Dependence of Mutation Rate on Radiation Intensity
60 River Surveys for Nuclear Power Plants
65 Recent Studies Related to Determination of MPC Values
66 References

68 VI CURRENT EVENTS

68 EBR-I Operating Experience
71 Idaho Chemical Processing Plant Criticality Incident of January 25, 1961
73 Comments on Proposed Reactor Site Criteria
76 Changes in Licensing Regulations
77 Action on Reactor Projects by Licensing and Regulating Bodies
83 Safeguards Reports and Selected Reading
83 References

Issued quarterly by the U. S. Atomic Energy Commission. Use of funds for printing this publication approved by the Director of the Bureau of the Budget on June 26, 1959.

Letter to the Editor

Hazards Analysis of Chemical Processing Facilities

In the recent article by Arnold and Nichols on "Hazards Analysis of Chemical Processing Facilities" (*Nuclear Safety*, Vol. 3, No. 1, September 1961), the equations for downwind dose do not explicitly state the atmospheric dilution factors or the effective stack height of the effluent. These factors are no doubt contained in the various numerical constants. However, if this is true, the authors should have pointed out that the atmospheric dilution factor is by no means constant and can vary by several orders of magnitude during the day, depending on the local meteorological condition. If the intent of the authors was to describe, for instance, "average daytime meteorological conditions," this should have been pointed out in the article. As the "downwind dose" equations now stand, the indicated values could vary approximately by factors of 10 less to 1000 greater, depending on existing weather conditions.

Assistance in determining the range over which dilution factors vary can be obtained by referring to USAEC Report WASH-740, Theoretical Possibilities and Consequences of Major Accidents in Large Nuclear Power Plants (March 1957). Also, Gifford's recent articles on atmospheric dispersion in the March 1960, December 1960, and June 1961 issues of *Nuclear Safety* give practical assistance in evaluating the dispersion "constants" under various meteorological conditions and for various downwind distances.

DONALD H. PACK, Chief
Environmental Meteorological
Research Project

ISAAC VAN DER HOVEN, Meteorologist
Environmental Meteorological
Research Project

Pack and Van der Hoven are correct in their inference that atmospheric dilution factors and effective stack height are contained in the numerical constants. It should have been pointed out that k, the stack dilution factor (seconds per cubic meter), is defined by $k = 2/(\pi e u h^2)$, where u = wind velocity in meters per second and h = effective stack height in meters. Thus this dilution factor is that for which the ground exposure is maximized. This dilution factor is derived in Chap. 4 of USAEC Report AECU-3066 by differentiating and maximizing certain of Sutton's results.

Although the review did not show numerical results, in the actual hazards analysis of ORNL chemical processing facilities, the authors used the maximum credible value of this dilution factor which is obtained by evaluation of k at a wind speed of 3 mph. Not only is this wind speed representative of conservative meteorological conditions, but also, as may be seen from the definition of k, at higher wind speeds the dilution factor is lower, and at lower wind speeds the dilution factor would also be lower if credit were taken for the height of the plume rise as described in Chap. 5 of AECU-3066. — *The Editor*

Section I

GENERAL SAFETY CRITERIA

Reactors, Sites, and Safety

By M. C. LEVERETT*

The subject of reactor safety has been a live one during the past year. A set of fairly definitive reactor location criteria has been issued.*[1] Our first fatal reactor accident has occurred.[2] The safety of a major reactor project has become a matter of litigation in our highest court.[3] The AEC has given new recognition to the importance of reactor safety by reorganizing and expanding its staff.[4] The intention here is to discuss this highly technical subject in a rather broad fashion, taking up three principal aspects of reactor safety: (1) the philosophy of reactor safety; (2) reactor siting criteria; and (3) a description of some technical matters in the field of reactor safety which need attention.

Safety Philosophy

The seemingly simple question "How safe do you want to be?" has as many answers as there are experts who have addressed themselves to the question. It is interesting to consider the following answers:

1. The answer "as safe as other industries" may be put on various bases. One basis is the usual one of frequency of lost-time accidents in disabling injuries per million man-hours. The atomic energy industry, with a record of about two such accidents per million man-hours, is already safer than many industries reckoned as adequately safe. Another basis is the probability that an "average reactor" will cause a death during a year, which may be computed to be about 1 chance in 75 (although the statistics are very poor). Comparing this, for example, with 1 in 26 for a railway locomotive, 1 in 6 for a transport aircraft, and 1 in 2000 for an individual automobile indicates that reactors are safer than a good many devices which we accept as being adequately safe.

*This article is a condensation of the address by American Nuclear Society (ANS) President M. C. Leverett at the June 4–8, 1961, meeting of the ANS in Pittsburgh. Dr. Leverett is Consulting Engineer, Hanford Atomic Products Operation, General Electric Company, Richland, Wash. He received the B.S. degree from Kansas State College in 1931 and the D. Sc. degree in chemical engineering at Massachusetts Institute of Technology in 1938. He worked for the Phillips Petroleum Company from 1933 to 1935 and joined the Humble Oil Company in 1938. On loan to the University of Chicago, he took part from 1942 to 1943 in the initial work on the Hanford reactors. In 1943 he transferred to the Clinton Laboratories at Oak Ridge, where he was in charge of engineering of the Materials Testing Reactor, which was later built in Idaho. In 1948 he returned to Humble Oil Company, but he left after one year to become Technical Director of the NEPA (nuclear aircraft propulsion) Project at Oak Ridge. In 1951 he transferred with the Project to the General Electric Company at Cincinnati, in charge of all technical activities. Since 1956 until earlier this year, he had been Manager of the Development Laboratories Section of the Aircraft Nuclear Propulsion Department. He is Chairman of American Standards Association's Nuclear Standards Board Project N6, "Reactor Safety Standards," and past Chairman of the ANS Nuclear Standards Board. He led the U. S. delegation at the meetings of the International Organization for Standardization on reactor safety standards in 1958 and 1960. He is past Vice President (1959–1960) and past President (1960–1961) of ANS. He is a Fellow of the American Nuclear Society and an Associate Fellow of the Institute of Aerospace Sciences. He is a member of A.I.Ch.E., A.I.M.M.P.E., A.P.S., and the Engineering Society of Cincinnati.

*See also the discussion in Sec. VI under the heading Comments on Proposed Reactor Site Criteria.

2. Another answer is that "no substantial increase in risk to individuals" should be caused. This criterion has been adopted by some responsible groups, including the Advisory Committee on Reactor Safeguards (ACRS). In the entire nuclear energy industry, the risk of accidental death due to all causes is about 1 in 40,000, as compared with the present risk to an individual of 1 in 2000 from all causes and of 1 in 7000 in the home. Certainly the nuclear energy industry is adequately safe by this criterion. A possible defect in this approach is the fact that none of the large-scale accidents which have been calculated to be possible have occurred in the nuclear industry. One such accident could change the statistics markedly.

3. The answer that "The probable damage over a long period should not exceed some set level" expresses an entirely different approach. For example, it has been suggested for a given industry that the public injury or damage which might be acceptable should be proportional to the contribution of that industry to the public welfare, possibly as measured by the dollar value of the services or products provided. This is a very objective approach and in a theoretical sense has considerable attractiveness. However, it involves estimating the probability that an accident will occur which is at present practically impossible either a priori or from experience. Use of such a criterion, therefore, is at present impractical; also, it is probably a little bit cold-blooded for general public acceptance.

4. A fourth type of answer which has received some publicity is: "No matter how safe it is; if you can make it safer, do so!" This criterion is at the opposite end of the objectivity scale from No. 3 above and is essentially emotional and unreasoned. It is unfortunate that some of this view has crept into our thinking with respect to safety in nuclear matters. Obviously, this kind of criterion is uneconomic and very difficult to apply since there is no logical limit to the extent to which one might be driven in the name of increasing safety.

A somewhat different approach is to observe that about 350 decisions have been made to build and operate various reactors in the United States, thereby accepting a certain level of risk. These have been thoughtful decisions, in some cases agonized over at length by many conscientious and informed people. If a pattern can be found in them and codified, the code can be used as a standard in judging future proposed reactors. The recently announced AEC "site selection guide"[1] is a good example of the emergence of such a pattern. Other examples of patterns will doubtless emerge on further study. Insofar as possible, such criteria should be expressed in terms of performance or objective to be achieved rather than in details of design or construction of a reactor. For example, a desirable type of criterion would specify the off-site dosage in event of an accident, together with needed assumptions in the calculation of such dosage. It would not simply specify, for example, a relation between reactor power and distance. It is important to remember that such standards are evolutionary developments based on many difficult considerations. The standards follow and are changed by practice instead of preceding and determining practice. Sizable departures may very well be proper in individual cases, thus providing more background for modification of the standards. This approach would leave room for reduction of cost, without compromising safety, through ingenious design; allow comparison with past practice, rather than with the credibility of an accident or with some absolute level of risk; offer a basis for relaxation from unnecessarily stringent and pessimistically based prior practice.

In summary it is proper that safety criteria be derived from practice instead of from "first principles"; be evolved with experience, being regarded as indicators rather than laws; and be meant to be departed from when the justification is suitable. This is the basis on which many of our "standards" have been developed—speed limits for automobiles, factors of safety in structural design, and standards of purity in milk, to mention a few. The use of such comparative criteria involves difficult problems, such as judging the relative probability and seriousness of accidents between two reactors of different design, but is easier than judgments of the absolute probabilities and intensities of the same accidents. In many cases a qualitative judgment will suffice if a quantitative one cannot be made.

Site Criteria

Decisions on the acceptability of a site for a reactor depend on many considerations. Much

attention has been given to site criteria in the past year, and there is reason to think that comparative judgments of the safety of sites will become much easier in the future.

The possibility and desirability of evolving an index of merit for sites has been considered. It has been the basic practice to judge the suitability of a reactor site by the probable damage a large accident might cause. There has been some discussion as to whether the index of damage should be human injury or property damage. Probably either is satisfactory as a relative index, but the absolute numbers resulting should not be taken too seriously. Too often the site evaluation is required to set forth the worst possible accident at a site, regardless of its probability. The compounding of improbable combinations can produce frightening damage numbers, but they tax one's credulity. Such calculations are useful only in a relative way and should not be used to determine the required dollar assets of the reactor owner, as sometimes appears to be done.

The proposed guide published by the AEC a few months ago for use in siting reactors essentially codifies existing siting practice.[1] The nuclear industry appears to consider the criteria to be in the main sound and workable, but with room for improvement. One weakness in the guides is that, if taken literally, they compensate only for possible total size of accident but not for the probability, which is logically unsupportable and surely not intended. Another weakness is that they do not allow for a realistic reduction of pressure in the containment vessel following an accident, a factor that can in many cases make an important difference in the maximum damage probability. Another difficulty is in the definition of population zones. This could be obviated by prescribing the total man-rem dose as the severity criterion, which would remove the risk of defining "population center" and would limit the total radiation injury to the population. Meanwhile, the guides should be accepted until something better appears, with whatever modifications may be made as a result of the recent intensive discussions of them by the industry.

Codification of siting standards in the United States should not imply, as some will object, that we are setting standards for other countries to follow. We may indeed, as necessity demands, tend toward the practice of other countries ourselves.

Desirable Safety Programs

It seems that neither industry nor government is doing as much as it should in the area of making the fruits of experience of others available to those working on reactor designs and hazards evaluation. There are four categories of things that ought to be done, as set out below.

1. More work should be done in the area of reducing the probability of accidents. Although the statistics of large reactor accidents are zero, there have been several small accidents, and even more near-accidents. Standards based on these and on other experiences must be evolved and, moreover, subscribed to. Better standards of design, operation, organization, or procedure could have prevented all the mishaps mentioned of which the author knows. The standards will have to be developed piecemeal, a little at a time, until we are able to write comprehensive standards on all types of reactors, operations, and circumstances. We must not fail to start simply because we cannot do a finished job now.

2. More work should be done in the reduction of severity of accidents. Ingenious and resourceful designers have come up with several devices for reducing the severity of an accident should it occur. Examples are pressure suppression and containment sprays. However, these are as yet only partially evaluated and, as a result, the designer does not know what credit he can take in terms of decreased severity of the maximum supposed accident or whether his evaulation of the effectiveness of his particular safety device is going to be acceptable. Other possibilities have not yet been evaluated. In addition, the entire process of escape of fission products from within the fuel to the outside environment needs more investigation. This includes more investigation of release from molten or overheated fuel elements, plate-out on surfaces, removal by condensing steam or water sprays, reduction of pressure with time, the fraction likely to escape even through an open (e.g., ruptured) pressure vessel, filtration of escaping gases and vapors, and removal by rainout in the immediate vicinity. Fortunately, there is an AEC program of this type, although it does not seem as aggressive as might be

desired. After a suitable series of small-scale tests and analyses, large-scale experiments should be carried out, perhaps culminating in intentional reactor runaways with measurements of fission-product release, deposition factors, and other parameters. The cost would probably not exceed that of the average reactor experiment program, and the results could be of great value.

3. Additional work should be performed on the evaluation of the magnitude of the risk involved in possible accidents. The magnitude of the risk of off-site damage is obviously a direct function of fission-product release, which has just been discussed. In addition, some work remains to be done in the meteorological field, such as determining reasonable probabilities for specific meteorological conditions and their durations at particular sites, rather than assuming either the worst conditions or improbably long durations.

4. More work should be done on the further study and evaluation of potential sources of accidents. Hazards reports frequently state that all conceivable accidents have been examined and that the particular reactor cannot experience an accident of significant severity. However, potentialities of accidents not previously envisioned turn up from time to time. An example is the ejection of a control rod from a reactor due to the pressure of water and steam within it, which could conceivably result in an excursion-generated shock wave. It would seem that experimental work on the resulting effects on the pressure vessel and other components of the reactor would be justified. Perhaps intentional chemical explosions, planned to simulate reactor excursions, could be used in test vessels. Shock might cause brittle failure of the vessel wall rather than ductile failure, with a resulting increase in accident severity. In addition, it has been found in tests of some alloys that the time to rupture under a given stress at high temperature may be reduced by up to several orders of magnitude if they have been subjected to irradiation and that this kind of radiation damage does not always anneal out at temperatures as high as 1200°F. At present there is almost a complete lack of data on the effects of radiation on high-temperature structural alloys, particularly in their long-term properties. The collection of such data is difficult and expensive but is important to reactor safety. It is understood that such a program is beginning.

Conclusions

We can now begin to see the general dimensions of the technical problems of reactor safety. Work classifiable as reactor safety research is presently going on in this country at the rate of about $12 million per year. Our total reactor development program is approximately $400 million per year. From the point of view of balance, it would not seem unreasonable that expenditures for reactor safety research should be stepped up. Perhaps for a few years a doubling of these expenditures would not be out of order. Such expenditures are trivial when compared with the estimated magnitude of the damages for a severe reactor accident. More realistically, such expenditures might produce very substantial savings (perhaps of the order of $10 million per year when we are constructing five new power plants annually) just in the shortening of distance between the generating-capacity and load centers. Additional less tangible savings are also likely. This work should be a part of the program of the AEC. Simultaneously, there must be the development of standards of reactor design, operation, organization, and procedure by industry that will assure safety without undue economic penalty. There must also be the collection of standard practices in the assessment of effects of nuclear accidents in reactors and the methods of calculating such effects. The total program will involve considerable radiation damage work, metallurgy, meteorology, and engineering analysis and experimentation. Some of the last should be essentially full-scale using specially constructed reactors and containment vessels. We need to move ahead in this program as rapidly as possible. (Condensed by L. A. Mann)

Nuclear Space Systems

Preparations are currently being made to fly a number of different space nuclear energy sources which range from the radioisotope-fueled sources of a few watts in power up to the kilomegawatt fission reactor for the nuclear rockets. Safety analyses of these nuclear sources follow the usual lines of reasoning that the general population should be protected from unreasonable hazards. In addition, since these

energy sources will fly over other than our own territory, an additional step of considering the diplomatic impact of the flight must be taken. It appears that initially each flight will require direct Presidential approval.

On June 28, 1961, a significant step in the utilization of nuclear energy sources in space was taken when the Navy's Transit IV-A navigational satellite went into orbit carrying a SNAP 3* radioisotope-powered auxiliary power unit. The flight had originally been prepared for June 6 following Presidential approval, but the State Department requested reconsideration because of the world situation at that time, and the flight was delayed for the additional period.

The general safety considerations in the use of nuclear energy sources in space were discussed in a previous Review[5] and at the Space-Nuclear Conference[6] of the American Rocket Society in May 1961. The periods considered were (1) the prelaunch phase, (2) the launch phase, and (3) the return phase. Prelaunch is defined as the period from construction of the source until liftoff from the launch pad. The launch phase proceeds from liftoff until a successful launching is completed. The return phase refers to that time, planned or unplanned, when part or all of the nuclear package returns to the vicinity of the earth. Branch and Conner[7] expect that the radioactivity which might be credibly released over the next 20 years from nuclear aerospace devices burned up in the upper atmosphere upon reentry, if distributed on the global scale, will cause only a minor change in the existing levels.

Radioisotope Energy Source

In general, the radioisotope systems present their greatest hazard during the prelaunch and launch periods, since at this time the isotopes are at their greatest power output of the entire mission. Some of the problems of radiological safety analysis for the 125-watt radioisotope-powered SNAP 1A have been discussed by Dick,[8] including core loading in a hot cell, transportation of the fueled core, transfer of the fueled core to the generator, generator shielding, launch abort procedures, vehicle failure on the launch pad, vehicle failure during ascent flight, and reentry from extraterrestrial flight. Safety analyses of all radioisotope energy sources follow this general outline, since these sources differ only with respect to power level and the specific isotope used.

The alpha emitters Po^{210}, Pu^{238}, and Cm^{242} are favored[9–11] for the small (few-watt) SNAP 3 sources since they have a high power density and present essentially no gamma shielding problem. The 125-watt SNAP 1A uses Ce^{144} and the accompanying Pr^{144} beta emitters and gamma emitters for a power source and requires a drainable mercury shield at the launch pad. The use of Sr^{90} or Ce^{144} in the larger units is dictated primarily by cost considerations. Both are produced from fission products.

The radioisotope is in each case tightly encapsulated in a high-temperature-resistant high-strength material capable of containing the materials under the conditions of a launch abort. The launch abort tests in the altitude range from 0 to 1000 ft have been described by Dick.[8] The extreme condition would occur following a booster failure in which the core could be subjected to a 6000°F fireball and explosion with shock-wave overpressures of 8000 psi as the booster exploded, followed by 8-min immersion in a high-temperature fire of from 2000 to 4500°F. Tests to approximate such conditions were made at Aberdeen Proving Ground on Inconel X, Hastelloy B, and type 316 stainless-steel cores. Inconel X was also tested by immersion in liquid oxygen to demonstrate low-temperature integrity and resistance to thermal shock. A study of abort conditions from 1000 ft to 300 miles indicated that the core would impact at 500 ft/sec and be at a temperature of 1500°F. Tests conducted against consolidated earth showed that the core would retain the radioisotope, but granite impact test results indicated that the core should be redesigned. Subsequent tests of redesigned cores with impact-absorbing caps gave satisfactory results.

As a general criterion for reentry, the requirement for complete burnup above 100,000 ft has been generally accepted. This means that the debris would be above the usual air travel lanes and would be retained for some time if the particle size were small enough. In reentry considerably more energy is liberated in a short time from the reentry drag heating

*The SNAP (Systems for Nuclear Auxiliary Power) units are given an odd number if they have a radioisotope energy source and an even number if they have a fission energy source.

than in the abort conditions, and it appears feasible to burn up the cores for both SNAP 1A and the various SNAP 3 units. Breakup of the core aids in the reentry burnup. Analysis of SNAP 1A indicated complete burnup to particles of sizes less than 10 μ above 120,000 ft.

Nuclear Reactor Energy Sources

The safety analyses of the SNAP reactor systems[12,13] as energy sources in space parallel closely the analyses of the radioisotope systems, except for the additional hazard of unplanned criticality. Shipment problems for the reactor are somewhat more difficult than for the radioisotope systems, although this problem is simplified when the fission-product inventory is kept low by limiting the preshipment testing of the reactor. Shipping provisions have to guard against accidental excursions of the reactor during shipment. The development of safety criteria and examination of hazardous conditions for the launch period are necessary for each of the SNAP reactors. Once at the launch pad, the integration of the power package with the rest of the payload is also made simpler if the fission-product inventory in the reactor has been kept low during the preflight period, since only minimum shielding of the reactor is necessary under those conditions.

If orbital startup of the reactor is used, the launch-phase problems will be minimized. Zero-power testing of the reactor can take place without an appreciable increase in the rate of fission-product inventory. If the reactor is operated at full power on the pad and during launch, the *gamma dose rates* will be within safe limits up to a thermal power of almost 500 kw. Remote removal of the reactor because of malfunctioning or long delay on the launch pad does not present difficult problems, but a reactor excursion on the launch pad or an abort followed by an excursion could present hazardous conditions. Analyses of the SNAP 2 [50-kw(t)] reactor indicate that a reactor excursion would release fission products equivalent to those present after operation for 30 min. Personnel evacuation radii and blockhouse protection appear adequate for radiological protection.

Reentry problems for the SNAP reactors can be minimized by placing the package in a long-lived orbit so that a long decay period is available after reactor shutdown. A 700-mile orbit which would allow the reactor about 300 years to decay would provide for safe, intact reentry. However, reentry analyses[14] of the fuel elements by P. D. Cohn have indicated that if the elements are released above 280,000 ft they will burn up and disperse above 100,000 ft. These analyses are still subject to experimental verification, and the long decay period is favored. Some combination of the two may be the ideal answer.

(P. G. Lafyatis)

JCAE Hearings on Radiation Safety and Regulation

The Joint Committee on Atomic Energy (JCAE) held public hearings in Washington June 12–15, 1961, on radiation safety and regulatory problems. The proceedings of these hearings are now available.[15]

(W. B. Cottrell)

Breeder Reactors and Nuclear Safety

The problems of controlling nuclear hazards in power reactors that breed do not differ in kind from those in nonbreeding power reactors, but they do differ in intensity as a result of the requirements for breeding. For the purposes of this review, a breeder reactor is defined as one that is capable of producing fuel in excess of its own requirements at the rate of not less than 4 per cent per year of the total inventory of fissile isotopes in the reactor system, including the fissile isotopes in processing and fabrication. Power systems based on breeder reactors, as just defined, have doubling times of 25 years or less and are thus suitable for use in an expanding nuclear power economy. Fertile isotopes would be the only nuclear raw material. The cost of any electric power produced is not relevant to this definition.

Excluded by this definition are breeder reactors having fuel yields smaller than 4 per cent per year; these are excluded because they are unsuitable for use in an expanding economy based solely on supplies of fertile isotopes. Also arbitrarily excluded are reactors that produce fissile isotopes different from those used for fuel; e.g., a reactor fueled with U^{235} and producing U^{233} in a blanket loaded with Th^{232}.

The fuel yield in a power-breeder reactor station is related to the system characteristics by the relation

$$Y = \frac{37.8GRFQ}{I} \quad (1)$$

where Y = yield of excess fuel, per cent per year
G = breeding gain, i.e., the net number of fissile atoms produced in excess of the reactor's own fuel requirements per reaction between a fissile atom and a neutron in the reactor
R = fuel consumption ratio, number of fissile atoms destroyed by reaction with neutrons per fission reaction, $= 1 + \alpha$
F = plant factor, average attained fraction of rated capability
Q = rated station thermal capability, thermal megawatts
I = station-associated inventory of fissile isotopes, in kilograms, including all fissile isotopes and their precursors (e.g., Pa^{233}) in the station and those prorated from associated processing and fabrication plants.

The designer of a breeder reactor is, of course, concerned with minimizing the net cost of power. Equation 1 constitutes a restraint on the ranges of the various independent variables at his disposal within which he may minimize the power cost. Over some of the factors in Eq. 1 he has little control; for example, in breeder reactors $R \sim 1.2$ and its value is more a function of the choice of fuel cycle (Pu vs. U^{233}) than of the design of the reactor. Also, the plant factor F is customarily taken as 0.8 in economic calculations, although the plant operator will, of course, attempt to achieve greater utilization in a base-load plant.

The definition of breeder reactors adopted above requires that the yield be not less than 4 per cent per year. With $R = 1.2$ and $F = 0.8$, and setting $Q/I = S$ [S = the specific power in Mw(t)/kg], the designer of a breeder reactor must choose a concept and select independent variables such that

$$GS \geq \sim 0.1 \quad (2)$$

The designers of breeder reactors have not found it easy to satisfy Eq. 2. Current conceptual designs, almost without exception, are based on optimistic assumptions and the extrapolation of current technology. In a recent study of thorium breeder reactors,[16] it was found that breeding gains lying in the range from 0.05 to 0.1 can be achieved in "practical" systems only if core materials having the very lowest parasitic cross sections are selected, amounts of structural materials are drastically limited, both fuel and fertile streams are processed rapidly, xenon is removed quantitatively from fluid-fuel reactors in less than 1 min, and excess reactivity is not wasted on parasitic control poisons. Obviously, these restrictions greatly complicate the problem of making the reactors safe.

With $0.05 \leq G \leq 0.1$, the specific power S must lie in the range from 1 to 2 Mw(t)/kg if Eq. 2 is to be satisfied. At first glance, this may not seem an extreme requirement; however, when allowance is made for the large inventories of fuel isotopes that accumulate in the blankets, processing plants, fuel reserves, etc., the specific power S_c for the fuel in the core becomes very large for some concepts; e.g., $S_c \sim 20$ Mw(t)/kg in a gas-cooled graphite-moderated thorium breeder.[16] In general, some parameter (e.g., power density, velocity, pressure and temperature rise, or heat flux) must be pushed near the limits of safe design if the required specific powers are to be achieved. Thermal reactor concepts for which G is less than 0.5 are probably not capable of satisfying Eq. 2.

Fast reactors characteristically show much larger breeding gains than thermal reactors, and core materials may be selected almost without regard to nuclear properties (except slowing-down power). Calculated values of G cited in the literature[17–21] are listed in Table I-1.

Table I-1 VALUES OF BREEDING GAIN FOR SOME FAST REACTORS

Reactor	Author and reference	Estimated breeding gain, G
EBR-II (plutonium-metal)	USAEC[17]	0.2
PFFBR (plutonium oxide)	APDA[18]	0.34
PuO_2-UO_2 fast-breeder	Greebler and Aline[19]	0.45
Plutonium-metal fast-breeder	Hummel[20]	0.59
PuC-UC fast-breeder	Sofer and Sapir[21]	0.57

(For some idealized systems containing no fission products,[22,23] the calculated G is greater than unity.) Table I-1 is based on estimates which take into account penalties for the presence in the core of fission products, as well as structural materials and coolants, and the

buildup of higher isotopes. Apparently none of the authors considered the effect of fissile isotopes and fission products in the blanket on leakage and other neutron losses.

It was concluded that "practical" fast reactors are probably capable of breeding gains of 0.5, in the limit, and therefore specific powers S of at least 0.2 Mw(t)/kg are required in order to satisfy Eq. 2. This may not be easy to achieve, however, because fast reactors characteristically have small cores, and the required power densities [Mw(t)/cu ft] become high. The Plutonium-Fueled Fast-Breeder Reactor (PFFBR) is expected[22] to produce 775 Mw(t) with an initial inventory of 674 kg of Pu^{239} in the core. When allowance is made for bred fuel in the blanket and fuel in processing and fabrication, the total inventory may well be two to four times as great, corresponding to specific powers of 0.3 to 0.6, which, taken with the corresponding G listed in Table I-1, barely satisfies Eq. 2. The required power density in the core is 13 Mw(t)/cu ft average and 25 Mw(t)/cu ft maximum. Obviously, this high concentration of heat sources poses problems of heat transfer in normal operation and of afterheat removal in case of loss of the coolant pumps. In general, however, since there is greater freedom in selection of core materials, it appears that the conditions in the fast reactors are somewhat less extreme, in relation to their limits, than those in most of the thermal breeders.

The cost of power from breeder reactor stations decreases with increasing station power, and proposed stations therefore tend to have high power; e.g., 1000 Mw(e) and above. In particular, since it is necessary to process the fuel and fertile materials frequently, the capacity of the processing plants must be high in order to reduce the unit costs. High-power rating alone poses problems in selecting an exclusion area and in estimating the maximum credible accident for multiple reactor stations.

In nonbreeding power reactors, nuclear hazards may always be reduced by compromising the performance, both nuclear and thermal. In power-breeder reactors, however, in order to satisfy the requirements for breeding, the choice of core materials is restricted, and the structural materials must be limited in amount. Power levels, specific powers, power densities, operating conditions, processing rates, etc., all must be made to approach upper limits. As a result, the difficulty of meeting nuclear safety requirements is greater than in systems where the performance may be compromised without regard to breeding requirements.

(L. G. Alexander)

Safety in Source Handling

The handling of source material has been discussed indirectly in previous *Nuclear Safety* reviews of related subjects,[24–27] but this is the first review specifically concerned with the safety aspects of source handling. For the purposes of this review, source is defined in sufficiently broad terms to include, in addition to sealed sources (as defined in the *Code of Federal Regulations*[28]), sources of radiation which can be generated at will but which are intended for the same use as sealed sources. Specifically, portable or fixed X-ray machines used for radiography or for research purposes and accelerators of all types are considered here to be radiation sources. The following general criteria for the control of radiation hazards at major nuclear installations at ORNL are generally applicable to sources:

1. The maximum credible accident must be contained or confined to the degree necessary to preclude the discharge of concentrations or amounts of radioactive materials which are injurious to health of affected persons or which will interfere with normal activities in the area.
2. Two lines of defense must be present to prevent the escape of radioactive materials. The two lines of defense must exist at all times, and adequate monitors must be provided to indicate when either fails.

In the actual review of a facility or operation to assess whether the safety criteria have been satisfied, two important questions should be asked after the details of the situation have been presented:

1. Has the individual responsible for the facility correctly evaluated the hazards presented by normal operation and by off-design operation (including the maximum credible accident) in relation to the precautions to be taken to minimize or eliminate such hazards?
2. Should the residual hazards be accepted as reasonable for the type of work, location, im-

portance, proximity of personnel, etc., involved in the operation of the facility?

The first of these questions emphasizes that the most important element of radiation safety is the human element. Individual ability and an individual sense of responsibility are required; no amount of safety equipment is believed to be able to cope with the consequences of irresponsible, incompetent, human action. In any safety review, therefore, both the facility and the individual in charge must be evaluated.

The philosophy of double containment with detectors between the two containers has required some modification for application to sources, as distinguished from reactors, radiochemical plants, hot-cell operations, hot-laboratory operations, waste disposal, and reactor experiments. It is recognized that most of the standard types of radiation sources are doubly encapsulated rather than doubly contained, and there is usually no substantial clearance between the inner and outer capsules. Once the outer capsule has been closed by welding, it is, of course, impossible to know whether the inner capsule has remained intact or has developed a leak. Therefore most AEC and commercial radiation sources cannot be said to be protected by two lines of defense with provisions for testing the adequacy of the first. In evaluating its position, ORNL management has stated[29] that "... the first encapsulation for sources is carefully checked by smearing to prove the integrity before the second encapsulation takes place. Therefore the probability of a leak is small. In addition, good practice entails periodic smearing of the outer encapsulation to assess the integrity of the source. These precautions are adequate, and we would not recommend a more elaborate system for proving the existence of double containment."

Nevertheless the importance of good containment—or, rather, the consequences of poor containment of sources—was forcibly demonstrated in connection with the 1957 incident in Texas involving iridium sources. The public relations aspects of the situation proved far more serious than the actual radiation damage involved (discussed in the September 1961 issue of *Nuclear Safety*, page 73). The unloading and loading of sources, even doubly contained, often requires violation of a primary containment barrier. Under such circumstances, administrative control must be exercised so that the secondary containment line is activated and tested before the primary containment is deliberately violated.

The following sections illustrate by comments or criteria some of the radiation safety precautions pertinent to the use of various types of sources. A recent incident that typifies a number of accidents involving sources is discussed in the September 1961 issue of *Nuclear Safety*, Sec VI.

Gamma-Ray Sources

High-Level Nonportable Gamma-Ray Sources. Inasmuch as the exposure to these sources could be lethal within a few minutes, strictest precautions must be observed in their handling. An extreme example of such a source is the 10,000-curie Co^{60} source at ORNL,[30] which is housed in a special apparatus[31] and contained in a special, locked room. The hazards connected with its operation were considered to be derived primarily from personal exposure within the room. Special attention was given to the mechanical details of the construction of the source container and the movable shields.

The radiation monitors and alarms used with this source are of two types: (1) continuously indicating automatic alarm monitors that are regularly checked for proper operation by a member of the Health-Physics Division and (2) standard radiation-measuring instruments that require manual setting or use. Weekly smear tests of the irradiation cavities are used to prove the absence of leakage. Individual dosimeters and "pocket screamers" are worn by the experimenter, and special pocket screamers are placed on a movable shield dolly to indicate the malfunctions of the apparatus. The automatic radiation alarm works on an isolated electrical circuit that is locked out so that it cannot be disconnected by an experimenter.

Strict administrative control is imposed in that the room is locked. The key must be obtained from the responsible individual, and the person entering the room must be accompanied by a qualified health physicist. The health physicist determines the radiation level in the room and checks all the radiation monitoring instrumentation, but he is not required to remain with the operator. The room door is never left unlocked when the room is unoc-

cupied. Furthermore, it has not yet been considered necessary to prohibit the use of the facility to a single individual working alone. Nevertheless the experimental room is connected to one of the building offices (in which a secretary is usually present) by means of an intercommunication system so that, in an emergency situation in which the experimenter is injured or cannot leave the room, assistance can be summoned. A change in the designation of the individual responsible for this facility would require a new review of the adequacy of the safety precautions.

Field Units and Portable Intermediate-Level Gamma-Ray Sources. Less danger of a lethal exposure in a short time but much more danger of exposure of unsuspecting or careless persons is presented by the intermediate-level gamma-ray source used in portable form or in field units for radiography or for process control. A recent report[32] describes the inspection of a commercially manufactured Ce^{137} radiation-density gauge that was recovered from the wreckage of a cannery which had been totally destroyed by fire. Although the primary container prevented the spread of radioactivity, an aluminum-plate label attached to the assembly was partly melted (but still legible). The source strength was such as to give a reading of 44 r at contact. The recommendation was made that warning labels on such sources be stamped on, or otherwise permanently made a part of, both the primary and secondary containers.

Another commercial source that presents interesting features of radiation safety is a 10-curie Co^{60} source that is used for field inspection of mechanical equipment. The source pellet is driven by compressed air from a lead container through a plastic tube to a shielded head where it provides the radiation for the desired autoradiographs. After the picture is taken, a flow of compressed air returns the pellet to its container. Although the company services the unit on some regular basis, the sources are not being inspected and smeared by the company. Regular tests by laboratory personnel would be advisable. The compressed air used to blow the source pellet around is released to the room without being filtered or monitored. This condition is potentially hazardous, and the use of a filtered vacuum rather than pressure would be more desirable. Both light signals and an ordinary survey meter are used to indicate that the pellet is outside its lead container and in the exposure position. When not in use, the source is kept locked and the key is kept in a safe place.

With reference to the previous discussion of double containment of sources, it is recognized that only single containment of this source is present at the time it is being used for an irradiation. Therefore it is essential to maintain the integrity of the source.

Low-Level or Calibration Gamma-Ray Sources. A number of low-level, portable gamma- or beta-ray sources for calibration of instruments are in use. Some of these have presented problems because of their small size and because of inadequate labeling. To prevent exposure to uninformed personnel, it is recommended that a permanent label be made a part of each source, that the sources be painted in the usual warning colors, and that strict accountability control be maintained by a system of written records and locked storage.

X-Ray Sources

The use of commercial X-ray equipment for radiographic, crystallographic, and biological studies is widespread. The following recommendations were advanced for their use at ORNL to focus attention on the special problems involved in the use of such equipment:

1. All X-ray machines should be registered with the applied health-physics groups.

2. Health-physics surveys should be made routinely during normal operation of the apparatus, or after any major relocations or changes in the apparatus, to establish the radiation exposure of the operators and the isodose contours around the apparatus. The routine surveys should be made at least once a year and the results recorded in duplicate, with one copy in the registry file for the machine and one copy posted either at the machine or at the entrance to the room in which it is kept.

3. Office personnel should be located as far from X-ray equipment as is necessary to prevent their accumulating low-dosage-rate exposures over long periods of time.

4. Research and development attention should be initiated on the problem of obtaining more accurate radiation exposure records for the fingers and hands of the operators. The develop-

ment of X-ray burns is well documented by the medical profession. Operators should request and be provided with finger rings or bracelets containing film of the proper sensitivity for measuring soft X rays rather than films for measuring exposures to beta radiation.

5. Too frequently it has been found that radiography apparatus is used in makeshift rooms located adjacent to the working areas of large numbers of people. Proper permanent facilities should be constructed when it becomes apparent that substantial amounts of radiographic testing are required.

6. In connection with the health-physics surveys of operating equipment, particular attention should be given to the high-voltage rectification systems which may emit substantial quantities of soft radiation.

7. A device should be invented which will essentially make the location of the X-ray beam visible to the operator when the beam is turned on; this is particularly appropriate for horizontal beams.

Accelerators as Sources

Three sources of radiation are important in connection with the operation of accelerators: (1) the primary beam, (2) the induced or scattered radiation from the target, and (3) the residual radiation from radionuclides formed in the target. In addition to these dangers, there is the important nonradiation hazard that is connected with the use of very high-voltage electricity. On the credit side of this situation, it should be noted that the operators of accelerators usually have a more intimate knowledge of the nature of the radiation involved than do most users of radiation. Nevertheless, a number of comments have been made which are particularly appropriate for situations in which one building houses more than one accelerator.

Scheduling. The uses of the machines should be planned so that the supervisors of the building or complex can know exactly what is expected to cause the presence of radiation at any time.

Background Surveys. It has been suggested that a better knowledge of the effects of one experimental machine on the radiation levels at other machines would be of assistance in facilitating the interpretation of local radiation levels.

Centralized Monitoring. There should be a single location where the results of all radiation monitoring are continuously displayed so that, together with the schedule, an over-all evaluation of the building and experiments can be quickly made.

Personnel Exposure. Interlocks are generally accepted for protection against personnel exposures, but they frequently cause inconvenience to the operators, particularly when adjustments on the machine and the location of the beam are being made. Television viewing of the interiors of the target rooms has been suggested as one safety feature for use in connection with strict administrative control where there is the need to enter while the beam is on. Panic buttons have been recommended which would permit an operator to shut off the beam from a position near the target. Radiation monitoring by a health physicist is recommended during beam adjustments because of the demonstrated difficulty of estimating time reliably when engaged in adjustments that require concentration. Additional attention to the development of accurate remote-control positioning equipment would permit many experiments to be performed with greater safety. Supersonic devices that would detect the presence of a person inside a target room have been suggested for consideration.

Work at Night and on Weekends. Work on off hours is accepted as normal by experimenters, and it is frequently difficult to enforce administrative safety controls. Assistance should be given to the experimenter by providing the easiest possible way for him to ensure that some responsible officer on the grounds knows of his presence in the building during these times.

Tritium Targets. Targets such as tritium represent an example of hazards not connected with the accelerator beam itself. Special precautions must be taken in the fabrication, storage, and disposition of the targets; in the operation of ventilation systems in areas where tritium might be released; in monitoring for tritium at the target area; and in monitoring experimenters for the possible exposure to tritium.

Presence of Untrained Personnel. Craftsmen are required for much of the work in connection

with accelerator experiments, and precautions should be taken that they are not permitted or required to enter unsafe areas. It is expected that thermonuclear experiments will create sources of considerable magnitude, complicated by the presence of very high energy neutrons which can interact with surrounding matter. Such sources are almost outside the definition used here and require special safety reviews because of their peculiar characteristics.

Alpha-Containing Sources

Possibly the most serious radiation hazard to uninformed or nontechnical people, including the general public, arises from the use of large amounts of alpha-emitting nuclides in radiation sources. The extremely low tolerance level for these materials is illustrated by the figures for Pu^{239} (accepted as the usual reference standard): the permissible body burden for Pu^{239} in bone tissue is 0.04 μc, and, since 16 g of this isotope produces 1 curie, the permissible body burden is 0.64 μg. Snyder has compiled a single criterion for determining the equivalent amounts of various nuclides on the basis of relative hazard.[33] A list of equivalent amounts based on this criterion is presented in Table I-2, as taken from an unpublished ORNL training manual.[34]

Table I-2 HAZARD EQUIVALENT TO 1 G of Pu^{239}

Radionuclide	Hazard equivalent	
	In grams	In microcuries
$_{84}Po^{210}$	2.7×10^{-6}	1.2×10^{4}
$_{90}Th^{230}$	0.78	1.5×10^{4}
$_{90}Th$ (natural)	5.5×10^{4}	6.1×10^{3}
$_{92}U^{233}$	7.3	6.9×10^{4}
$_{92}U^{234}$	11	6.7×10^{4}
$_{92}U^{236}$	1.2×10^{3}	7.6×10^{4}
$_{92}U^{238}$	2.2×10^{5}	7.5×10^{4}
$_{92}U$ (natural)	1.1×10^{5}	3.6×10^{4}
$_{93}Np^{237}$	80	5.5×10^{4}
$_{94}Pu^{240}$	0.27	6.1×10^{4}
$_{94}Pu^{241}$	0.011	1.2×10^{6}
$_{94}Pu^{242}$	16	6.1×10^{4}
$_{95}Am^{241}$	0.019	6.0×10^{4}

There is quite widespread use of neutron sources containing intimate mixtures of alpha emitters and materials such as beryllium with which alpha particles interact to produce neutrons. Fortunately, most of these are quite ruggedly made and are doubly encapsulated in welded metal containers. As was indicated above, such care is taken in their construction, including smearing to establish the integrity of the first capsule, that they can be considered safe if regular smearing of the outside of the second capsule fails to reveal any leakage of activity. Naturally, adequate labeling and accountability practices are necessary to prevent their being handled by unauthorized persons. Also, it is expected that their use will be monitored by health-physics surveys to show the exact hazard offered by the neutron and gamma emission from these sources. In the past these sources have frequently been encased in plastic, but this is no longer a recommended procedure. No plastic or organic material is tolerated inside the welded containers in order to avoid the production of high pressures through alpha-induced decomposition of the organic material.

An even more serious hazard, although much less widespread, is presented by the recent development of Am^{241} gamma-ray sources; the Am^{241} 60-kv gamma ray is very useful for the determination of the concentrations of high-Z materials in process streams. Unfortunately, very thin windows must be used in these sources to minimize the attenuation of the weak gamma rays. Typical sources of this type consist of $\frac{1}{4}$ g of Am^{241} in the form of a thin layer or pressed pellet of the oxide contained in a stainless-steel capsule having a machined 10- or 20-mil-thick window at the end. The americium is deposited on the inside of the window. Although these sources are really quite rugged, they do present considerable hazard in that such a large equivalent amount of alpha material is contained by only one thin metal window. The users of these sources are furnished with safe handling instructions and information on the construction of the source. Nevertheless these sources have been furnished to Eurochemic, as well as to AEC and Defense Department agencies. There may be a question as to whether the potential hazard is such as to require even more stringent precautions in connection with their distribution and use than those currently in force.

Research personnel have continuing need for open alpha sources, such as Po^{210} plated on a metal disk. The hazards involved in such uses are so great that each use should be treated as an individual case. No general distribution of such sources seems to be tolerable.

Radiation Incidents Involving Sources

The completeness of this review has been facilitated by the recent publication of an AEC compilation of case histories of accidents involving radiation sources in nuclear energy operations.[35] These cases were tabulated in the June 1961 issue of *Nuclear Safety*, together with all other radiation incidents.[36] Included in this list were over 20 source incidents involving exposures to more than 50 persons.

Conclusions

The incidents, and the near-misses which have come to the attention of the reviewer, point out the very great importance of the human factor in the safe handling of radiation sources. The increased emphasis being given to radiation safety training by the AEC and by its contractors should provide a basis for an improved safety record in the next review period. Stricter controls over field radiography by AEC licensees would seem to be in order. (H. F. McDuffie)

References

1. Notice of Proposed Guide, Reactor Site Criteria, *Federal Register*, Feb. 11, 1961; reviewed in *Nuclear Safety*, 2(4): 1–3 (June 1961).
2. C. A. Nelson et al., SL-1 Investigation Board's Report; SL-1 Incident, National Reactor Test Site (undated). Transmitted by C. A. Nelson to A. R. Leudecke by attachment dated May 10, 1961.
3. *Atomic Energy Clearing House*, Special Issue (June 12, 1961).
4. AEC Separation of Regulatory Functions from Operation and Development Functions, *Nuclear Safety*, 3(1): 92–95 (September 1961).
5. Hazards of Nuclear Power Units in Space Applications, *Nuclear Safety*, 1(4): 1–3 (June 1960).
6. P. G. Lafyatis, Hazards Associated with Nuclear Power Reactors in Space Applications, Paper No. 1754-61, American Rocket Society–Oak Ridge National Laboratory, Space-Nuclear Conference, Gatlinburg, Tenn., May 3–5, 1961.
7. I. L. Branch and J. A. Connor, Jr., Nuclear Safety in Space, *Nucleonics*, 19(4): 64–68 (1961).
8. P. J. Dick, Safety Analyses and Tests of a Radioisotope-Powered Thermoelectric Generator, Paper No. 1341-60, American Rocket Society, Space Power Systems Conference, Santa Monica, Calif., Sept. 27–30, 1960, reported in *Progr. Astronautics and Rocketry*, 4: 547–561 (1961).
9. W. Hagis and G. P. Dix, Final Safety Analysis Report, SNAP 3, Thermoelectric Generator, USAEC Report MND-P-2364, Martin Co., Nuclear Div., June 1960.
10. D. Knighton, Preliminary Operational Safety Report for the Task 5.6 Thermoelectric Generator, USAEC Report MND-P-2356, Martin Co., Nuclear Div., May 1960.
11. C. O. Riggs, Curium Fueled Generators for Lunar and Space Missions, USAEC Report MND-P-2366, Martin Co., Nuclear Div., June 1960.
12. F. D. Anderson and J. G. Lundholm, Jr., Evaluation of SNAP Safety for Space Reactor Application, Paper No. 1335-60, American Rocket Society, Space Power Systems Conference, Santa Monica, Calif., Sept. 27–30, 1960, in *Progr. Astronautics and Rocketry*, 4: 351–363 (1961).
13. F. D. Anderson and M. Huntsiger, SNAP Space Reactor Flight Safety, Nucleonics in Space Symposium, Dallas, Tex., Mar. 28, 1961.
14. P. D. Cohn, Paper No. 1336-60, The Experimental and Analytical Programs for Reentry Burnup of SNAP Reactors, American Rocket Society Space Power Systems Conference, Santa Monica, Calif., Sept. 27–30, 1960.
15. *Radiation Safety and Regulation, Hearings Before the Joint Committee on Atomic Energy, June 12–15, 1961*, U. S. Government Printing Office, Washington, 1961.
16. L. G. Alexander et al., Thorium Breeder Reactor Evaluation. Part I. Fuel Yield and Fuel Cycle Costs in Five Thermal Reactors, USAEC Report CF-61-3-9(App.), Oak Ridge National Laboratory, May 24, 1961.
17. Civilian Power Reactor Program, Part III. Status Report on Fast Reactors as of 1959, USAEC Report TID-8518(1), Book 1, Jan. 1960.
18. Atomic Power Development Associates, Inc., Fast Reactor Core Design Parameter Study, USAEC Report APDA-133, March 1960.
19. P. Greebler and P. Aline, Parametric Analysis of a PuO_2-UO_2 Fueled Fast Reactor, in Proceedings of the Conference on the Physics of Breeding, Oct. 19–21, 1959, USAEC Report ANL-6122, p. 116, Argonne National Laboratory, October 1959.
20. H. H. Hummel, Breeding in Fast Plutonium Metal Systems with EBR-II Type Recycle, in Proceedings of the Conference on the Physics of Breeding, Oct. 19–21, 1959, USAEC Report ANL-6122, p. 137, Argonne National Laboratory, October 1959.
21. G. A. Sofer and J. Sapir, The Potential of Carbide Fuels in Fast Breeder Reactors, *Trans. Am. Nuclear Soc.*, 3(1): 85 (June 1960).
22. S. Yiftah and D. Okrent, Some Physics Calculations on the Performance of Large Fast Breeder Power Reactors, USAEC Report ANL-6212, Argonne National Laboratory, December 1960.
23. W. H. Roach, Computational Survey of Idealized Fast Breeder Reactors, *Nuclear Sci. and Eng.*, 8(6): 621–651 (December 1960).

24. Basic Considerations of Radiation Protection, *Nuclear Safety*, 2(1): 14–18 (September 1960).
25. Safety in Hot Laboratory Design and Operation, *Nuclear Safety*, 2(1): 51–53 (September 1960).
26. Plutonium Handling, *Nuclear Safety*, 1(3): 49–52 (March 1960).
27. Organization for Standards in the Nuclear Field, *Nuclear Safety*, 1(3): 13–21 (March 1960).
28. *Code of Federal Regulations*, Title 10, Part 30, Radioisotope Distribution; see also *Federal Register*, pp. 2136–2139, Mar. 15, 1960.
29. F. R. Bruce, ORNL Internal Memorandum, Oct. 28, 1960.
30. W. Davis, Jr., Chemical Technology Division 10,000-Curie Co^{60} Source; Description, Operating Procedures, and Safety Procedures, USAEC Report CF-60-3-82, Oak Ridge National Laboratory, March 1960.
31. J. A. Ghormley and C. J. Hochanadel, A Cobalt Gamma-Ray Source Used for Studies in Radiation Chemistry, *Rev. Sci. Instr.*, 22: 473–475 (1951).
32. P. W. Howe et al., Inspection of a Fire-Damaged Radiation Source, USAEC Report UCRL-9291, University of California Lawrence Radiation Laboratory, June 29, 1960.
33. W. S. Snyder, A Single Criterion for Determining an Equivalent Amount of Various Radio-Nuclides on the Basis of Relative Hazard, USAEC Report CF-60-3-116, Oak Ridge National Laboratory, Mar. 23, 1960.
34. Oak Ridge National Laboratory, 1961. (Unpublished)
35. *Atomic Energy Clearing House*, 6(50): 19 (Dec. 12, 1960).
36. Accidents in Nuclear Energy Operations, *Nuclear Safety*, 2(4): 55–63 (June 1961).

Section II

ACCIDENT ANALYSIS

Fission-Product Release from UO_2

A major problem associated with fuel elements containing bulk UO_2 is the release of volatile fission products from the UO_2. In a previous review article,[1] it was pointed out that two important areas of uncertainty in predicting fission-gas-release rates are the temperature distribution in the UO_2 and the rate of release of volatile fission products from a given batch of UO_2 at a given temperature. Since the publication of the last review article on fission-product release from UO_2, several papers have appeared which help to reduce these uncertainties.

Temperature Distribution in UO_2

Postirradiation examination of some of the earliest UO_2 fuel elements tested revealed that regions of columnar grains were present at the centers of the fuel rods. Since these structures were reminiscent of the structure of metal castings, it was concluded that the UO_2 had been molten at the center of the fuel rod during irradiation and that the observed structure developed upon solidification. By assuming that the edge of the columnar region was at the UO_2 melting point, 5000°F, Eichenberg et al.[2] estimated the average thermal conductivity of irradiated UO_2 between 750°F and the melting point to be 1 Btu/(hr)(ft)(°F). This value has been used widely in the design of reactors with UO_2 fuel elements.

Further studies revealed that the centers of fuel rods where columnar grain growth had occurred had probably not been molten after all. Robertson[3] observed that outside the columnar grains there was a narrow band of relatively large, equiaxed grains. Outside these, the structure of the pellets was unchanged. From a knowledge of the grain-growth characteristics of UO_2, an approximate temperature was assigned to the outer limit of the equiaxed-grain region. An average thermal conductivity of the UO_2 was then computed assuming that the edge of the columnar-grain region was at the melting point of UO_2. An improbably low value for the thermal conductivity was obtained.

In a series of out-of-pile tests, MacEwan and Lawson[4] and Newkirk and Bates[5] duplicated the columnar-grain-growth region without central melting and postulated the following mechanism of columnar grain growth. In a steep temperature gradient at temperatures near 3600°F, where the vapor pressure of UO_2 is appreciable, UO_2 sublimes on the hot side of small pores and condenses on the cold side. The pores then move toward the center of the rod and leave behind a single, oriented grain. Volatile fission products in the path of a pore are collected in the pore and released to the central void. Subsequently pores at the periphery of the columnar region move through the same region and continually sweep out the volatile fission products in the affected region.

Additional confirmation of this mechanism was provided from examination by Robertson et al.[6] of fuel elements irradiated at much higher heat ratings than those of the earlier tests. It was observed that, within the region of columnar grain growth, there was another sharply outlined region. After short-time irradiations (less than 1 min), this innermost region was observed, but the region of columnar grain growth was greatly reduced. Thus it was believed that the inner region, whose extent was not time dependent, was the molten zone. By assigning a melting point of 5070°F to the boundary of the innermost region and esti-

mating the surface temperature to be 480°F, Robertson et al.[6] calculated that

$$\int_{480°F}^{5070°F} k(\theta)\, d\theta = 75 \text{ watts/cm}$$

This corresponds to an average thermal conductivity between 750°F and the melting point of about 1.7 Btu/(hr)(ft)(°F) for UO_2. Below 1800°F, the thermal conductivity values of Robertson et al.[6] were in agreement with the data of Scott.[7] Above 1800°F, Robertson's results indicated a constant value of 1.62 Btu/(hr)(ft)(°F) for the thermal conductivity of UO_2. More recent data[8] indicate that the thermal conductivity at high temperatures may increase slightly.

More recently, Bates[9] has published evidence that the thermal conductivity of UO_2 increases with increasing temperature above about 2000°F. This evidence was based on postirradiation examination of an element operated at a high heat flux and the correlation of the observed structures with the known temperatures at which equiaxed grain growth and columnar grain growth occur.

From postirradiation examination and data on unirradiated UO_2, Bates gave the following equation for the thermal conductivity of UO_2:

$$k = \frac{54}{18 + T}\frac{\rho}{\rho_0} + 4.37 \times 10^{-13}\, T^3 \qquad (1)$$

where k = thermal conductivity, Btu/(hr)(ft)(°F)
ρ = fuel density, g/cm^3
ρ_0 = UO_2 theoretical density, g/cm^3
T = absolute temperature, °R

The first term in Eq. 1 is in accord with the normally observed decrease in thermal conductivity with increasing temperature for ionic solids. The second term is the contribution arising from the radiant heat transfer in porous solids. The agreement between observed and calculated values confirms the postulate that radiant heat transfer causes an enhancement of UO_2 thermal conductivity at high temperatures. Equation 1 predicts a value of 2.1 Btu/(hr)(ft)(°F) for the average thermal conductivity of UO_2 between 750 and 5000°F.

Although the discrepancy in the results of Bates[9] and Robertson et al.[6] is unresolved, it is clear that the average thermal conductivity of UO_2 is higher than the 1 Btu/(hr)(ft)(°F) previously supposed. Further, the discrepancy is minor if the fuel-element heat ratings are modest and the central temperature of the UO_2 is not above 3000°F.

The above data were obtained with pressed and sintered UO_2 pellets and probably do not apply for the recently developed swaged and vibratory-compacted fuel elements, at least when they are operated at temperatures below the sintering temperature. The recent preliminary data of Lamartine[10] indicate that the thermal conductivity of swaged UO_2 is about one-half that of pressed and sintered pellets with the same density.

Rates of Release of Fission Products

Numerous investigators[11–20] have established that the primary mechanisms of fission-gas release from UO_2 are diffusion and recoil below about 2700°F. The amount released by recoil is a function of the shape and density of the fuel body, but it is independent of temperature. Under the normal operating conditions of most fuel elements, the recoil release is negligible compared with the amount released by diffusion. Thus most studies have been concerned with the rate of diffusion of fission products in UO_2. Above 2700 to 2900°F, the processes of grain growth, columnar grain growth, and sublimation occur. The effects of these processes have not been assessed quantitatively, but they may account for the abnormally high rates of release observed while structural changes were occurring.[14,16]

A comprehensive review of work on fission-product release from UO_2 prior to the fall of 1959 was published by Cottrell et al.[21] A compilation of values of the diffusion coefficients of fission products reported by various investigators was given, as were the equations for predicting the fractions released under various conditions. The diffusion coefficients for inert gases in UO_2 showed considerable uncertainty, even in regions where the equations appeared to be applicable. The scatter in the data was about five orders of magnitude at 2550°F. Reported values of the activation energy[13] were between 34 and 81 kcal/mole. In view of these uncertainties, Cottrell et al.[21] concluded that any particular fuel material should be characterized by a measurement of the diffusion rate constant, $D' = D/a^2$, where a is the radius of a sphere whose volume is that of the average solid fuel particle. The quantity D' arises naturally in the

solution of the diffusion equation in spherical geometry and can also be used with success to characterize release from nonspherical particles. The wide scatter in D' values at a given temperature for UO_2 fabricated by different processes was concluded to result from lack of understanding of the effects of important parameters affecting release, such as thermal conductivity, density, distribution of open and closed pores, grain growth, oxygen-to-uranium ratio, material discontinuities, and temperature structure.

More recent investigations have cleared up a number of these uncertainties. Long et al.[16] have shown by neutron-activation studies that there is good correlation between the diffusion rate constant and the Brunauer-Emmett-Teller (BET) surface area in the range 1500 to 2900°F. The UO_2 compacts varied in surface area from 7 to about 23,000 cm^2/g. Over this wide range, the following equation was found to apply:

$$\log D'_{2550°F}\,(sec^{-1}) = -13.55 + 2 \log \text{S.A.} \quad (cm^2/g)$$

where S.A. is the BET surface area.

From D' values and surface areas, the diffusion coefficient D was calculated and found to follow the equation

$$D\,(cm^2/sec) = 7.8 \times 10^{-6} \exp(-71{,}000/RT)$$

where T is in °K, in the temperature range 1500 to 2900°F. In the temperature range 2900 to 4000°F, Long et al.[16] observed that the rate of evolution of fission products was higher than that expected from diffusion alone and that the slope of the fraction released versus the square root of time increased with time. The effect was attributed to grain growth, which was also observed to occur in this temperature range.

Independent studies of Toner and Scott[14] on the release of Xe^{133} from UO_2 using the neutron-activation technique confirmed the results of Long et al.,[16] and the following relation between the BET surface area and D' was found to apply:

$$\log D'_{2550°F}\ (sec^{-1}) = -14.0 + 2 \log \text{S.A.} \quad (cm^2/g)$$

Above 2900°F the rate of release was higher than would be predicted on the basis of diffusion. This was attributed to sublimation of UO_2 and grain growth. Contrary to early work by Eichenberg et al.,[2] no correlation between D' values and UO_2 density in the range 90 to 96 per cent of the theoretical density was found. The differences between the results of the two investigations can probably be explained as differences in the samples used; all the measurements of Eichenberg et al.[2] were made on pellets fabricated by a single method, whereas the data of Scott and Toner were obtained on pellets fabricated by a number of commercial vendors and by a variety of methods. In the density range studied, the ratios of the BET surface area to the total surface area varied by two orders of magnitude for pressed and sintered pellets of the same density but fabricated by different techniques.

Stevens et al.[15] have observed that a correlation exists between UO_2 density in the range 93 to 97 per cent of the theoretical density and D' values for pellets made by a given method but that there are exceptions to the general behavior. These exceptions have been related to three microstructures which were anomalous for the density concerned. Stevens et al.[15] also found evidence which suggests that the initial high rate of release normally observed during neutron-activation tests[11,14,16] is due to an oxygen-rich surface layer in which the xenon diffusion rate is much higher than in the bulk of the oxide.

As a result of the recent work, the scatter in the values of the diffusion coefficient for Xe^{133} in UO_2 has been considerably reduced from the five orders of magnitude reported by Cottrell et al.[21] to about 1.5 orders of magnitude. A compilation of recent values obtained at different laboratories is shown in Table II-1. The reasons for the remaining scatter are not known.

Table II-1 VALUES OF THE DIFFUSION COEFFICIENT FOR Xe^{133} IN UO_2 AT 2550°F

Laboratory	D, cm^2/sec	Type of UO_2	Ref.
ORNL	7.5×10^{-16}	Sintered	14
Harwell	4.0×10^{-15}	Sintered	16
BMI	1.1×10^{-16}	Fused	19
Chalk River	5.1×10^{-16}	Fused	15

Values of D at 2550°F may be extrapolated to other temperatures by use of the Arrhenius equation once the activation energy is established. An average value of 65 kcal/mole was used by Cottrell et al. in making calculations. More recently, Long[16] reported a value of 71 ± 5

kcal/mole. Since this latter value was obtained in a hydrogen atmosphere, the effects of oxygen on the activation energy should have been negligible. Toner and Scott[14] and Stevens et al.[15] observed a range of activation energies for different types of UO_2. The former work was done in a vacuum, and the latter was done in helium; therefore trace amounts of oxygen may have affected these results. Further work is needed to clarify present uncertainties.

As discussed previously, Long et al.[16] and Toner and Scott[14] observed that rates of release were higher than those predicted on the basis of diffusion above about 2900°F. Until further information is obtained on the mechanisms responsible for the effect, it seems safest to anticipate that all the fission products are released from those portions of the UO_2 which are above 2900°F.

Studies of Fission-Gas Release from UO_2 During Irradiation

Two important limitations of the neutron-activation technique for studying fission-gas release are that the burnups are low and the effects of fissioning cannot be evaluated. These limitations are removed when the rates of release are measured in in-pile capsules through which a carrier gas is passed. By monitoring the amounts of short-lived krypton and xenon isotopes collected in suitable traps, much information can be obtained on rates and mechanisms of release. A major difficulty with such sweep capsules is a temperature limitation set by the construction materials. Additional problems arise in temperature measurement and control, flux monitoring, and maintenance of the purity of the sweep gas.

The rates of release of fission products from UO_2 in sweep capsules have been measured by Carroll[20] and by Melehan and Rough.[19] Additional studies are being made by Stubbs and Webster[22] at Harwell. Carroll's measurements were made over the temperature range 800 to 1800°F. In the range 800 to 1000°F, Carroll observed the temperature-independent release that is characteristic of recoils. At higher temperatures the rates of release increased markedly with increasing temperature in a manner characteristic of diffusion. The activation energy for the diffusion of Kr^{88} from UO_2 was found to be 30 kcal/mole. It was also observed that iodine diffused in UO_2 at about the same rate as xenon and krypton and had the same activation energy.

The low-activation energy suggested that the UO_2 was oxidized during the test. This was confirmed by postirradiation examination, as well as by observations of the release of stored energy from the UO_2. Carroll concluded that trace amounts of oxygen in the helium sweep gas were sufficient to oxidize the UO_2 during the test period. Oxidation was eliminated in later tests by the use of 3 per cent H_2–97 per cent He as the sweep gas. In both atmospheres, Carroll observed bursts of activity on heating and cooling of the sample.

In a second series of tests,[23] Carroll and Baumann used platelets of ThO_2 outside the UO_2 to trap recoils. Since the diffusion rates of volatile fission products in ThO_2 could be expected to be about the same as in UO_2, it was hoped that the ThO_2 would eliminate direct recoils. However, results showed no significant differences between the recoil contributions in the presence and absence of the ThO_2; this observation suggests that the "recoil" contribution does not come solely from direct recoils.

Melehan and Rough[19] helped to clarify this point with studies of two types of UO_2 in the temperature range 500 to 1500°F. One sample was a single crystal of fused oxide, and the other was a pressed and sintered platelet of UO_2. The surface areas for direct recoil (geometric surfaces) were the same in the two samples; however, the BET surface area of the pressed and sintered material was about 100 times that of the fused-oxide sample. In the temperature range studied, only a very small temperature dependence of the release was observed, but surprisingly the rates of release were found to be proportional to the total surface area and not the geometric area.

Melehan and Rough[19] explained the anomalous release in these low-temperature experiments in terms of a knockout mechanism originally proposed by Lewis.[24] According to this theory, gas release occurs by localized evaporation of the fuel surfaces during the passage of fission recoil fragments near the surface. The energies of these vaporized fission gases would be much lower than primary recoils and thus would not reenter the fuel. Since some of these fission gases would be retrapped by the condensing UO_2, the amounts of fission gases would be dependent on the efficiency of recapture. However, in neither the work of Carroll[20,23] nor

that of Melehan and Rough[19] is there any evidence that the diffusion of volatile fission products is greatly increased by simultaneous fissioning or that there is any increase in release which could be directly attributed to burnup.

The work with sweep capsules leads to the conclusion that a new mechanism of release has been demonstrated. However, the temperature range studied has been too limited for use of the data to predict the contribution of this mechanism to the release of fission products in fuel elements that normally operate at considerably higher central temperatures.

A limited amount of information is available on the release of fission products from UO_2 at high burnups. On the basis of data obtained from in-pile capsules, Toner and Scott[14] concluded that there was no marked effect of burnup on D' values up to a burnup of 16,400 Mwd/metric ton. At higher burnups at temperatures below 1500°F, however, Bleiberg et al.[25] found that the crystal structure of UO_2 was destroyed and that a large fraction of the fission gas was released. The effect was found to be initiated at a burnup of 60,000 Mwd/metric ton. Such an effect is not surprising in view of the change in composition associated with this burnup.

Estimates of Fission-Product Release in Fuel Capsules

A relatively simple method for predicting the amounts of stable fission gases released in fuel elements was given by Cottrell et al.[21] It was proposed that the temperature distribution could be computed by conventional heat-transfer equations. An average temperature is then computed by use of the equation

$$\bar{T} = \frac{2T_{\text{max.}} + T_{\text{surface}}}{3} \qquad (2)$$

A D' value associated with $\bar{T}$ is then selected on the basis of neutron-activation test results, and the fraction released is calculated by use of the equation

$$F = \frac{4}{\sqrt{\pi}} \sqrt{D't} \qquad (3)$$

where t is the irradiation time. Equation 3 applies for low values of the fraction released and when the structure of the UO_2 is unchanged by irradiation. For a given set of D' values, the fractions computed by the above method were found to agree well with fractions computed by summing the release from a number of incremental volumes in the fuel elements.

Scott[26] used the method described above to estimate the fractional release of Kr^{85} from prototype fuel elements for the Experimental Gas-Cooled Reactor (EGCR). These elements were irradiated in the Oak Ridge National Laboratory Research Reactor (ORR), and actual fractions released were measured by postirradiation puncture tests at the Vallecitos Atomic Laboratory of the General Electric Company. Maximum burnups were of the order of 2000 Mwd/metric ton. In making estimates a value of 1.62 Btu/(hr)(ft)(°F) for the thermal conductivity of UO_2 in the temperature range 1300 to 2900°F was used, and D' values were calculated from measured BET surface areas by using the neutron-activation test results of Toner and Scott.[14] The data were analyzed for the case of a 3-mil helium-filled gap between fuel and cladding and for the case of no gap. The results showed that the measured values of the Kr^{85} released generally fell within or close to the limits set by the 3-mil helium-filled gap and no gap. Further, most of the measured values were close to those predicted for a 3-mil helium-filled gap when the temperature of the type 304 stainless-steel cladding was about 700°C. For those elements having cladding temperatures of about 800°C, the measured release values corresponded more closely to the condition of no gap. These results are reasonable since the external pressure on the capsules was 320 psia and since creep with subsequent cladding collapse would be more likely to occur at the higher temperatures.

Robertson[27] used a similar method of analysis to predict the dependence of fission-gas release on heat flux. These predictions were tested for a series of capsules which were irradiated for the same time at different heat fluxes and which contained specimens from the same batch of sintered UO_2. Predicted values were compared with the amounts of free gas collected by puncturing the cladding after irradiation in an evacuated enclosure. Close agreement between predicted and observed release was found. Robertson[27] asserts that "these tests and their interpretation have shown that it is unnecessary to assume anything like 100% release in even the most highly rated [high heat flux] fuel ele-

ments currently designed." He also remarks that "designing for 100% release constitutes a deplorable waste."

Concluding Remarks

Although much work is still needed to clarify mechanisms of fission-gas release from UO_2, reasonable estimates may now be made of the amounts released in in-pile capsules, at least for pressed and sintered UO_2 at modest heat fluxes. The major areas of uncertainty requiring further investigation are the fuel-cladding interaction and the in-pile behavior and thermal conductivities of nonsintered UO_2 (e.g., vibratory-compacted and swaged oxides). Emphasis should also be placed on the long-term effects of structural changes on release in fuel elements operating at high heat fluxes with both sintered and nonsintered oxides.

(D. F. Toner and J. L. Scott)

Activity Buildup in Pressurized-Water Reactor Systems

The corrosion of in-core metals and the transport of "crud" by the coolant in a pressurized-water system results in some buildup of activity throughout the primary system. As experience has been gained with operating reactors, it has been observed that this activity buildup produces radiation fields that may be a safety hazard to personnel. The radiation levels encountered may only limit accessibility during routine operation, but they may also pose more serious problems during extensive maintenance operations. This article brings up to date a previous article on this subject.[28]

Activity Buildup

The term crud is used to designate the corrosion and wear products produced in the circulating system. Studies of the behavior of the crud indicate that a portion of it continuously migrates from one part of the system to another. At any particular time, only very small amounts (<1 ppm) of dissolved and suspended material are present in the circulating coolant,[29] but the rapid accumulation of a large number of coolant cycles results in effective transport. In the course of this transport, the crud is in neutron-flux regions a portion of the cycle time. Deposition in high-flux regions has been attributed to two mechanisms,[30] a hydraulic effect that operates mainly on the particulate matter and radiation effects that operate on colloidal or soluble matter. The activated nuclides that reenter the coolant stream are distributed through the crud of the system by both particulate- and ion-exchange processes that are usually irreversible.[29] Table II-2 lists some of the important radioactive nuclides that have been found in the circulating streams of pressurized-water reactors.[31]

Table II-2 REACTIONS IN COOLANT THAT PRODUCE ACTIVATION DURING NORMAL REACTOR OPERATION

Reaction	Radiation	Half life
$Fe^{58}(n,\gamma)Fe^{59}$	Beta, gamma	45 days
$Ni^{58}(n,p)Co^{58}$	Gamma	71 days
$Co^{59}(n,\gamma)Co^{60}$	Beta, gamma	5.2 years
$Cr^{50}(n,\gamma)Cr^{51}$	Gamma	27 days
$Fe^{54}(n,p)Mn^{54}$	Gamma	300 days
$Ta^{181}(n,\gamma)Ta^{182}$	Gamma	112 days

The importance of the various nuclides in crud activity buildup is indicated by data given in Table II-3 for the Shippingport Pressurized-Water Reactor (PWR) and the Stationary Medium Power Plant No. 1 (SM-1)[29] (formerly the Army Package Power Reactor).

Table II-3 INDUCED ACTIVITIES FOUND IN PRESSURIZED-WATER-REACTOR CRUD

	Activity, dis/(min)(mg of crud)		Crud activity as percentage of total activity	
Nuclide	PWR	SM-1	PWR	SM-1
Co^{60}	5.8×10^6	3.9×10^5	39.2	10.8
Co^{58}	2.5×10^6	21.0×10^5	16.9	58.0
Fe^{59}	1.8×10^6	5.0×10^5	12.1	13.8
Cr^{51}	2.2×10^6	4.9×10^5	14.8	13.5
Mn^{54}	0.8×10^6	1.4×10^5	5.4	3.9

The distribution of the activity among the several nuclides is, of course, dependent on the specific reactor. Thus the relatively high Co^{60} activity in the PWR crud shown in Table II-3 is attributed to the fact that the control-rod mechanisms are located above the reactor core.[32] As a result, corrosion and wear products from Stellite (~50 per cent Co) control-rod components may drop down and lodge in

high-flux regions with resultant high production of activated cobalt. Similarly, the relatively high quantity of Co^{58} in SM-1 is attributed to the fact that the SM-1 fast-to-thermal flux ratio is seven times that of the PWR; this, combined with the stainless-steel cladding, favors the high energy (n,p) reaction[29] on Ni^{58} in the SM-1 relative to the PWR, which uses Zircaloy cladding.

The buildup of activity and its nature, then, are a function of many variables, including primary system materials, core materials, reactor characteristics, coolant chemistry, primary system flow, and primary system geometry. Some of these variables may apply generally to reactors of given type, whereas others may be specific to a given reactor, with characteristic changes in the activity buildup encountered. In general, of course, the long-lived activities are the most important since they will predominate after long periods of power production. Fission-product activity has not been found to be a significant contributor to activity buildup so far.[29]

The radiation levels that result from buildup of activated crud range from 0.1 r/hr or less to several roentgens per hour. Cutie Pie measurements of the SM-1 primary system[29] gave the following radiation levels:

1. An elbow in the pressurizer line reached 4 r/hr, but the level dropped to 50 mr/hr when the system was drained. The geometry of the elbow was such as to provide a trap for loose material which presumably flushed out.

2. Primary piping radiation levels ranged from 0.2 to 0.6 r/hr, depending on geometry factors.

3. The impeller and shroud of one of the 4000 gal/min primary coolant pumps read 2 r/hr at contact.

4. The radiation levels were 0.4 and 1.4 r/hr on the vaporizer and superheater sides of the steam generator, respectively. This compares with a level of 2 r/hr "as close as practical" to the tube sheet of a PWR steam generator.

5. The dose-rate buildup at all points was approximated by the equation $y = cx^2$, where y is the dose rate in milliroentgens per hour, x is the number of months since reactor startup, and c is a constant ranging from 0.2 to 1.0, depending on the particular point in the system.

Similarly, the work-area radiation field during removal of the rod-drive mechanisms of the PWR in the first refueling operation was 150 to 400 mr/hr with shielding and 1 r/hr without shielding. Individual components of the mechanisms ranged in activity level from a few tenths to a few roentgens per hour as a result of deposited crud.[32]

The presence of such radiation fields results in an appreciable increase in the hazards to personnel. This is particularly true during maintenance operations when the plant is, of necessity, in an abnormal condition. The radiation levels and possibilities for spread of contamination during such periods require the application of stringent measures for the protection of personnel. Health-physics supervision of radiation dosages, protective clothing, and equipment, as well as facilities for decontamination and active waste disposal, are needed, not only for safety but also for efficient work progress. Thus there is material incentive for the reduction of such buildup from both the safety and economic viewpoints.

Activity Control

As experience with operating reactors has directed attention to the importance of this phenomenon, studies have been undertaken to provide means of controlling it. Some of the important means indicated by such studies are discussed briefly below.

Design Features. The most obvious means of reducing the crud concentration in the coolant is use of the following design features:

1. Elimination of crud traps and accumulation areas, such as cracks, crevices, and points of abrupt change in flow.

2. Minimization of moving-part wear and location of such parts so that wear products are least likely to get into the coolant stream.

3. Wherever possible, use of materials which are not a source of undesirable nuclides, e.g., minimize cobalt content of stainless-steel fuel cladding[33] and substitute noncobalt-containing alloys, particularly in the core region.

The mathematical equations developed by several investigators[34,35] are useful in developing special materials specifications.

Coolant Chemistry. Experience in operating reactors indicates[36] that activity buildup is reduced when the coolant pH is raised from 7 to 10. This is attributed to a reduction in trans-

portable crud as a result of lower corrosion rates and the formation of more tenacious films at the higher *p*H's. Lithium hydroxide is the recommended reagent for *p*H control for a variety of reasons: it does not cause radiolytic dissociation, it is the least aggressive reagent with respect to caustic stress-corrosion cracking, and there is a considerable backlog of prooftesting of it in operating reactors. Careful exclusion of oxygen from the coolant system has also been demonstrated to be important, presumably because increased corrosion due to the presence of oxygen results in higher crud levels.[37]

Purification System. Fixed-bed mixed-ion exchangers are conventionally used for coolant purification; however, in the KAPL-37 Test Facility and the S3G Reactor,[34] their effectiveness is rather low (0.10 to 0.15) when judged on the basis of the ratio of the activity removed by the purifier to the activity of the out-of-flux system. Other purification means, including evaporation, hydrocloning, and centrifugation, have been examined,[38] but they were discarded on the basis of higher cost. It is conceivable, however, that another system, such as evaporation, which could give very high decontamination[39] factors for all constituents of the recycled water, might give lower over-all costs when activity buildup costs are completely evaluated.

Decontamination. Considerable effort has gone into the development and testing of decontamination reagents and techniques. Several reagents and techniques have been satisfactorily tested in the laboratory and in loops, but the experience on reactor systems is more limited.

(E. G. Bohlmann)

Studies in Reactor Kinetics

One of the fundamental problems involved in designing a reactor is to determine the response of the system to a large, fast reactivity increase. In research reactors many types of accidents might result in such reactivity insertions; for example, void collapses in or near the core, fuel elements dropping into or onto a nearly critical reactor, poison samples falling out, and others. Even in power reactors which are not normally subject to such vagaries, "cold slugging" or loading accidents may conceivably cause such an excursion in some cases. The designer therefore needs to know (1) what the reactivity effect of a given configuration change will be, (2) what positive period a given reactivity step will give rise to, and (3) what the system "defense," either inherent (void and temperature coefficients, etc.) or imposed (control rods, fuses, etc.), will achieve in limiting the excursion.

The relation between the change in the neutron economy and the period such a change occasions in the reactor is embodied in the so-called "inhour equation" which relates a reactivity, ρ, with a period, T. Gross and Marable[40] have shown that the inhour equation can be rigorously derived by considering gross neutron conservation in terms of a "dynamic" multiplication factor, k_d, or alternatively in terms of the more commonly used "static" multiplication factor, k_s. The static reactivity, ρ_s, is defined as

$$\rho_s = \frac{k_s - 1}{k_s}$$

and the dynamic reactivity is equivalently defined in terms of the dynamic multiplication factor, k_d. The space-independent inhour equation for the dynamic reactivity is given by

$$\rho_d = \frac{\Lambda}{T} + \sum_i \frac{\beta_i}{1 + \lambda_i T} \qquad (4)$$

where Λ is the neutron generation time, β_i and λ_i are the ith delayed-neutron absolute yields and precursor decay constants, respectively, and T is the stable reactor period. Under certain conditions, which in practice are applicable in almost all cases to thermal reactors, it is shown that, similarly, the static reactivity may be stated as

$$\rho_s = \frac{\Lambda}{T} + \sum_i \frac{\beta i(\text{eff})}{1 + \lambda_i T} = \frac{\Lambda}{T} + \gamma \sum_i \frac{\beta i}{1 + \lambda_i T} \qquad (5)$$

where $\beta i(\text{eff}) = \gamma \beta_i$ is the effective delayed-neutron fraction which takes account of the relatively increased effectiveness, γ, of delayed neutrons in causing further fissions due to their lower initial energy and hence increased nonleakage probability. The approximation is made here that the nonleakage probability of all delayed-neutron groups is the same, i.e., that γ is a constant applicable to all i delayed-neutron groups.

Fundamentally, the measurements of changes in reactor multiplication are measurements of periods, which are then converted to reactivities by equations such as 4 and 5 above in order to attain additivity of the effects of small changes. To interpret results of such measurements, it would then be necessary to know which definition of ρ was employed. However, a far more "natural" method is available for reporting such data. If both sides of the above equations are divided by

$$\beta = \sum_i \beta_i \quad \text{or} \quad \beta_{\text{eff}} = \sum_i \beta_{i(\text{eff})}$$

respectively, one obtains the relation

$$\rho(\$) = \frac{\Lambda^*}{T} + \sum_i \frac{\beta_i/\beta}{1 + \lambda_i T} \qquad (6)$$

where the reactivity is expressed in dollars and $\Lambda^* = \Lambda/\beta$ or $\Lambda/\beta_{\text{eff}}$ as the case may be.

The advantages of this method are several. For one, the ambiguity as to which reactivity definition is used is avoided. More important, in "static" measurements of reactivity where only the second term on the right enters the equation $[\Lambda^*/T \ll \Sigma_i(\beta_i/\beta)/(1 + \lambda_i T)$ for large $T]$, only the relative yields of the delayed-neutron groups enter, and these are known more accurately than the absolute yields. In "dynamic" measurements, however, such as those made by the pulsed-neutron method, only the first term enters, and it is apparent that the quantity Λ^* is the one directly related to the measured period. In fact, at criticality, Λ^* is just equal to the prompt period. With reactivity results reported in dollars, comparisons with calculations can be made unambiguously, since the quantity calculated depends on the calculational method and can be properly converted into dollar units based on a knowledge of the calculation.

Reactivity Measurements by Static Methods

As has been pointed out above, if reactivity steps are small, the "prompt" term of the inhour equation is negligible compared to the delayed-neutron terms, and knowledge is required only of the delayed-neutron precursor periods and relative yields to obtain the reactivity in dollars from period measurements.

The most common experimental technique, usually referred to as the inhour method, consists of making the reactor supercritical in successive small steps, measuring the stable positive period due to each step, and compensating for each step in turn by some other method, such as insertion of other control rods, uniform poison addition, and reflector-height variation.

A great deal of such work has been done. A particular experiment of this type, directly applicable to reactor safety, is the measurement by Reynolds et al.[41] of the reactivity effect of large voids adjacent to a pool reactor. Large air voids were constructed by moving water-filled Plexiglas containers inside an air-filled tank covering one entire face of a pool type research reactor. Changes in the reactivity resulting from variation in the size and location of the void and from varying the separation of the void from the core were measured by the inhour method using strips of distributed poison to extend the range of reactivity available. In this way reactivity effects of about 4 dollars were observed. Statistical weight correlations were compared with the experimental data, and the agreement was quite good, even though the voids were treated as gray absorbers of the same interface area, neglecting effective transport and energy change of neutrons reflected back to the core from void surfaces. The results are good enough to permit calculation of arbitrarily shaped and positioned voids in similar reactors by using the statistical weight concept.

Disadvantages of measuring the reactivity changes by the inhour method with "quasi-distributed" poison are (1) the long waiting time necessary for the asymptotic period to become established, with attendant possibilities for errors introduced by insufficient waiting time and (2) the possibility that errors may be introduced by flux-shape perturbations due to removal of some of the poison strips, and by the configuration change which is to be measured.

The first problem is avoided by a method discussed by Braffort[42] and Stubbs[43] and reported in application by Sastre.[44] In this technique the time variation in the neutron density is fed as input to an analog computer that solves the space-independent kinetic equations continuously in real time and thus affords an instantaneous reading of reactivity without awaiting the establishment of the asymptotic period.

In the experiment reported, the kinetic behavior of a heavy-water-moderated critical experiment was modeled by using six delayed-neutron groups and six delayed-photoneutron groups. This type of reactivity measurement, although offering the advantage of instantaneous information output, does have inherent difficulties. The major difficulty is due to the local nature of the neutron detector. Thus it responds not only to flux changes due to the reactivity of the reactor but also to local flux-shape changes caused by the configuration change. This effect is not a problem in period measurements, since the slope of the asymptotic flux does not depend on the change of flux magnitude at the detector caused by the configuration change. In the Sastre[44] method, after a long enough time, the reading on the reactivity meter will also record the correct reactivity again, but the time constant of this asymptotic approach to the correct value is the same as that for the inhour method. Thus, if it is necessary to wait for the asymptotic approach, the advantage of the Sastre method is negated. However, if the reactivity change is caused by uniform changes, such as temperature or uniform void-fraction changes, or if the distance from the perturbation to the detector is sufficiently large, the effects of flux perturbation may be negligible and a correct value of reactivity immediately following a change will result. The high expense of this scheme makes it attractive only in situations where many reactivity measurements are required.

Two other releated techniques are the "rod-drop" and "source-jerk" methods of determining reactivity changes. The essence of the rod-drop technique is to perform the reactivity change by inserting the negative reactivity into the initially critical reactor and observing the time change of the flux. In the source-jerk method the reactor is initially subcritical by the amount to be measured with a source maintaining constant flux; the source is rapidly removed, and the reactivity is determined from the transient response of the flux to the source removal.

Two techniques for interpreting the data obtained in such experiments have been described by Hogan.[45] Neither technique requires measurement of the flux as a function of time during the transient response. One method is an integral counting method; the other is a differential method.

The integral counting method involves summing all the neutron counts detected during the transient, which requires from 5 to 10 min for reactivities of the order of 1 dollar and longer times for smaller reactivity differences. (Note that Eq. 8 of Hogan's paper,[45] which gives ρ in terms of the measured quantities, is in error; the quantities γ_i should be replaced by λ_i.) The differential method requires a measurement of the linearly rising asymptotic flux resulting from the presence of a source in the critical reactor and a measurement of the magnitude of the level flux with the same source, after the rod-drop has been performed.

With the rod-drop and source-jerk techniques, as with the Sastre method, it is important to place the detector so that it does not "see" the spatial variation of the flux engendered by the perturbation.[46] The equations for the reactivity given by Hogan are for the static reactivity and can be transformed into equations for reactivity in dollars to remove the necessity for this distinction. The "integral method" equation then becomes

$$-\rho(\$) = \frac{N(0)}{n(0)} \sum_{i=1}^{6} \frac{\beta_i/\beta}{\lambda_i}$$

where $N(0)$ and $n(0)$ are, respectively, the count rate just after the perturbation and the total number of counts during the transient.

The prompt term can be included in this method, although, as Hogan points out, it is usually negligible. Including this term gives the relation

$$-\rho(\$) = \frac{N(0)}{n(0)} \left(\Lambda^* + \sum \frac{\beta_i/\beta}{\lambda_i}\right)$$

Similarly the exact reactivity equation in dollar terms for the "differential method" is

$$-\rho(\$) = \frac{\dot{N}}{N_1} \left(\Lambda^* + \sum \frac{\beta_i/\beta}{\lambda_i}\right)$$

where N and N_1 are, respectively, the linear slope of the flux in the critical reactor with the source present and the neutron flux density just after the source jerk.

Measurements of Prompt Generation Time

In the methods outlined so far, the prompt term in the reactivity equation has been either

completely negligible or has appeared only as a small correction term to the delayed-neutron terms.

However, the "prompt" term in the reactivity equation, Eq. 6, has a fundamental importance in the context of this review for two basic reasons. For one, there exists a class of reactivity measurements, namely, the "pulsed-neutron-source" measurements,[47–50] which relies exclusively on the measurement of prompt effects and thus depends only on the knowledge of the "reduced generation time" Λ^* and not at all on the knowledge of any delayed-neutron parameters. For another, this reduced generation time is the essential parameter which determines the response of the reactor to a large, fast reactivity increase and thus is the key to the excursion behavior of the system which determines the requirements on the safety system. Lewin[51] has pointed out that two neutron-lifetime parameters may be defined. The generation time Λ (which is defined by Gross and Marable in Eq. 30 of reference 40, but which they call the lifetime l^*) is the reciprocal of the neutron production per neutron and per unit time. The neutron lifetime l is given by the neutron destruction (leakage plus absorption) per unit time and per neutron (Gross and Marable call this the lifetime l^* also, as defined in Eq. 43 of reference 40). The relation between the two is $\Lambda = l/k_d$. Lewin shows that in a critical reactor the two are identical, but, if the reactor is not critical, they differ. In particular, he points out that a reactivity change made by a black absorber, which leaves the production unchanged, changes l, but leaves Λ unaltered; on the other hand, if a reactivity change is made with a gray rod replacing fuel so as to keep the absorption unaltered, then l remains unchanged but Λ is altered. These statements do not strictly hold, since the spectrum and the importance function are changed even if only a black absorber is moved or if the thermal absorption can be kept exactly unchanged.

The reduced generation time Λ^* can be measured by a variety of methods, including the pulsed-neutron technique, pile-oscillator and pile-noise analysis methods, static $1/v$-absorber method, and analysis of excursion experiments.

The pulsed-neutron technique has been applied in a number of experiments by deSaussure and Silver.[52–54] The reactor in a known condition of reactivity is pulsed with a short burst of neutrons, and the decay is observed during a time interval after the burst that is long enough for the fundamental spatial mode to have become established and so short that no delayed neutrons are observed. Under these conditions the first term in Eq. 6 is the only one entering, and we have the relation

$$\rho = \frac{\Lambda^*}{T_p} + 1$$

The theoretically simplest and most accurate way to know the reactivty is to have the reactor critical. Then Λ^* is just equal to the prompt pulse-decay period T_p. However, in this situation the pulsed source (representing an average constant source over long times) gives rise to a large linearly rising background against which the decay must be observed. This makes the accurate determination of Λ^* difficult.

The background is much reduced and remains constant in time, if the experiment is performed in a subcritical reactor. However, this entails two sources of error. For one, the dollar value of the reactivity must be known by some other measurement (usually inhour methods in a supercritical system with a different configuration from that of the subcritical pulsed system), and, for another, the value of Λ^* may be different in the subcritical configuration than at critical. So long as the difference from criticality is less than about a dollar, the latter effect is small, but, particularly if the reactivity change is due to production changes (as pointed out by Lewin[51]), a change in Λ^* will occur. The best pulsed-neutron values for Λ^* were therefore obtained by pulsing the critical reactor.

Spano[55] reports on both a static measurement using a $1/v$ absorber and an excursion analysis of Λ^*, both measured on the Bulk Shielding Reactor II (BSR-II), a light-water-moderated stainless-steel research reactor.[56] The absorber technique is based on the fact that the effect of a small uniform addition of a $1/v$ absorber can be written in the form

$$\frac{\delta k_s}{k_s} = -\frac{\delta \Sigma_a}{\Sigma_a + DB^2}$$

which, by the definition of l given by Lewin,[51]

is $-\delta\Sigma_a v_{th} l$. If a $1/v$ absorber of known cross section is used to obtain $\delta\Sigma_a$, then

$$\delta\Sigma_a = \frac{\sigma_{a0} v_0 N}{v_{th}}$$

where N is the atomic density of the added absorber if the reactor is bare and homogeneous. If the reactor is not homogeneous and bare, then N_{eff}, a properly defined effective mean absorber density, is used instead.

The experiment was performed by separately poisoning portions of the reflector or the entire reactor tank with dissolved boron. In this fashion the effectiveness integrals entering N_{eff} were evaluated, and a value of l^* was obtained by making inhour measurements, in dollars, of the reactivity effect of the poison addition. Since this value of l^* is obtained in a critical reactor under static conditions, $\Lambda^* = l^*$ in this case. The result, $l^* = 2.8 \pm 0.13$ msec, agrees with the value measured for this reactor by other methods. In the same report,[55] Spano discusses a determination of l^* by analysis of the excursion experiments on the BSR-II. The reciprocal of the stable positive period following a step insertion of a known amount of reactivity (by use of a control rod calibrated in dollars by the inhour method) is plotted against the quantity

$$(\rho - \rho_d)(1 + \rho\beta\gamma)$$

where ρ and ρ_d are the reactivity and the delayed portion of the reactivity corresponding to a given period.

The delayed effect is not experimentally avoidable here, as in the pulsed experiments, so that the ρ_d term must be calculated (using the Keepin et al.[57] delayed-neutron data) and subtracted. The resulting curve, covering a range from 50 down to 3 msec, is linear and passes through the origin within the limits of experimental accuracy. The slope of this line gives $l^* = 2.87 \pm 0.02$ msec, which agrees well with the static value obtained in the same core.

The pile-noise-analysis method for obtaining Λ^* (equal to l^* since the technique is used on a critical reactor) has been described in two papers by Cohn.[58,59] It is closely related to the measurement of l^* by the pile-oscillator technique. In both cases the parameter required is the square of the magnitude of the zero-power transfer function at high frequencies.

The zero-power reactor transfer function $G(\omega)$ is given by

$$G(\omega) = \frac{1 - j\omega l \dfrac{\beta_{\text{eff}}}{\beta} \displaystyle\sum_i \frac{\beta_i/l}{\lambda_i + j\omega}}{j\omega l \left(1 + \dfrac{\beta_{\text{eff}}}{\beta} \displaystyle\sum_i \dfrac{\beta_i/l}{\lambda_i + j\omega}\right)}$$

At high enough frequencies the λ_i becomes negligible compared to $j\omega$, where ω is the angular frequency, l the prompt neutron lifetime, and β_i and λ_i are the delayed-neutron yields and periods, respectively.

The equation then reduces to

$$G(\omega) = \frac{1 - \beta_{\text{eff}}}{\beta_{\text{eff}} + j\omega l}$$

The square of the magnitude is

$$|G(\omega)|^2 = \frac{(1 - \beta_{\text{eff}})^2}{l^2 \left[\left(\dfrac{\beta_{\text{eff}}}{l}\right)^2 + \omega^2\right]} = K \frac{1}{\left(\dfrac{\beta_{\text{eff}}}{l}\right)^2 + \omega^2}$$

In noise-spectrum measurements the noise-frequency spectrum is considered to consist of a white-noise contribution from instrument and detection statistics and a component from the pile-flux fluctuations. The mean-square amplitude thus has the form $A + [B/(\alpha^2 + \omega^2)]$, where $\alpha = \beta_{\text{eff}}/l$. The frequency response of the analyzer and detector system is measured with a white-noise source; the resulting curve shape for $|G(\omega)|^2$ versus ω can be fitted to the best value of the parameter α. Cohn reports such measurements in the ZPR-V reactor and four other reactors at Argonne National Laboratory, with results for each that are accurate to about 2 per cent in the best case but only 50 per cent in the worst, a D_2O reactor for which α^{-1} had a value of 20.4 sec^{-1}. This poor accuracy was due to the fact that the analyzer response rolled off very rapidly in the frequency region where the transfer function rolloff occurs. A frequency analyzer with a lower limit in frequency would improve these results.

The pile-noise-analysis method has also been applied to the BSR-II with a result[60] of $\Lambda^* = 2.6 \pm 0.3$ msec. This is not as accurate a measurement as was reported for the same reactor by the two other methods above, but it agrees within the limits of error.

Both transfer-function techniques have been used in analyses of the KEWB and the Sodium

Reactor Experiment (SRE) reactors, as reported by Griffin et al.[61,62] In the SRE a value of 5.25×10^{-4} sec was obtained with both techniques, but with errors of 6.7 per cent using an oscillator and 13.3 per cent by noise analysis.

With a pile oscillator the perturbations are not noise induced but, rather, are artificially inserted with some periodic absorber motion that is carefully designed to approximate a pure sine-wave function in reactivity. In this type of experiment, the amplitude and phase of the transfer function are directly observed, and the analysis follows the general lines described above. A careful study of oscillator design was carried out by Wasserman and Spano[63] for measurements in pool type reactors at the Special Power Excursion Reactor Test (SPERT) facility at the National Reactor Testing Station.

Dynamic Reactivity Measurements

The pulsed-neutron technique, which has been described above as a means of measuring Λ^*, is also applicable to the measurement of reactivity effects, with particular usefulness in cases where the flux-shape changes are not negligible and where the effects are of large magnitude (several dollars). Consideration of the "prompt reactivity equation"

$$\rho(\$) = \frac{\Lambda^*}{T_p} + 1$$

shows that, once Λ^* is known, the reactivity of a given reactor configuration, if it is negative, may be directly obtained from a measurement of T_p, the decay constant of the neutron flux in the reactor after a burst at times short compared to the delayed-neutron precursor periods, provided Λ^* remains constant or its change can be calculated. A good example of the application of this technique is the measurement of the reactivity worth of a central fuel element in the BSR-I by deSaussure et al.[54] This experiment, like the other pulsed-neutron experiments on the BSR-I and BSR-II reactors, was performed by mounting a 300-kv Cockcroft-Walton deuteron accelerator above the Bulk Shielding Facility pool, with a 23-ft deuteron flight tube leading down into the pool so that the tritium target was positioned in close proximity to the reactor core.

After determining the value of Λ^* in the critical reactor, the central fuel element was removed and the reactor pulsed to determine T_p. The report[54] is notable for its careful consideration of the change in Λ^* caused by the removal of the central fuel element. A perturbation calculation was performed to determine the change in Λ^*, using real and adjoint fluxes calculated by two-group PDQ code calculations, and flux integrals obtained from these using the APCO-1 code.[64,65] The central fuel element was found to have a reactivity worth of 6.1 dollars, which caused a change of 10.2 per cent in Λ^*. It is clear that changes of this order of magnitude are not negligible, and such calculated corrections to Λ^* in measuring multidollar reactivity effects with the pulsed-neutron method should always be made.

A somewhat different technique for making reactivity measurements with pulsed neutrons is the pulsed-neutron investigation of "models" of the reactor containing no fuel but having similar neutron diffusion characteristics. The decay constant of the thermal-neutron population in such a passive assembly with and without the mockup of the control rod present gives directly the change in the buckling due to the poison addition or other configuration change.[66,67] Such studies have been carried out with a cadmium absorber in water by Nilsson and Sjöstrand[68] and with variously shaped cadmium absorbers in beryllium assemblies by deSaussure et al.[69] This technique is useful in that it permits accurate calculational comparisons with control-rod calculations for black rods in thermal reactors where no epithermal effects are present, but, by the same token, the method has shortcomings as a device for obtaining reactivity worths of actual reactor control-rod mockups because these effects are neglected.

Measurements of Effective Delayed-Neutron Fraction

As has been repeatedly stressed above, knowledge of β_{eff} is not required in reactivity and kinetic work if the measurements are reported in dollar units and the reduced generation times are used. However, measurements of β_{eff} have been performed on several reactor systems and will be reviewed briefly. The method used for such measurements is the substitution technique first described by Ruane et al.,[70] wherein a small amount of fuel is replaced by an absorber of equal absorption cross section. Kaplan and Henry[71] have reported such

a measurement, with the analysis based on considering the substitution as an effective change in ν, the number of neutrons per fission. The poison used in this experiment was a mixture of boron and hafnium intended to mock up the absorption of the displaced fuel both in the thermal and resonance energy ranges. Experimental considerations made it necessary to make the displacement of fuel by the absorber in one small region at a time; the additivity assumed in this procedure is considered and justified. However, it adds to the uncertainty of the result. A final value is quoted as $\gamma = \beta_{eff}/\beta = 1.33 \pm 0.066$. This corresponds to $\beta_{eff} = 0.0085 \pm 0.0005$, using the β measured by Keepin et al.[57] The reactor used was a water-moderated critical assembly (submarine project critical assembly) with fuel elements consisting of zirconium and uranium strips held in aluminum clips.

The same experimental technique was used[72] by Perez-Belles et al. in the BSR-I. They consider the substitution to be a reactivity change which can be measured in dollars and calculated accurately in fractional units so that the ratio gives β_{eff} directly. It appears to the reviewer that this theoretical approach is more reasonable than the one employed by Kaplan and Henry, although both lead to the same results. In this case the fuel was replaced by B_4C dispersed in aluminum, one fuel plate at a time; the absorption cross-section mismatch was corrected by a calculation using real and adjoint flux integrals obtained from a three-group PDQ calculation. In this case the result in β_{eff} is given as 0.0080 ± 0.0002, or $\gamma = 1.25 \pm 0.05$. A direct calculation of γ was also performed, using the PDQ code calculations, which yielded a value of 1.25, in good agreement with the measurement. It is pointed out that Reynolds et al.[41] calculated γ using the relative nonleakage probabilities in an equivalent bare BSR-I configuration, with a result of $\gamma = 1.22$, also in agreement with the measurement, so at least in thermal, light-water-moderated reactors, a simple nonleakage probability calculation based on the Fermi age to thermal of fission and delayed neutrons would appear to be adequate for estimating γ.

Reactor Excursions

The SPERT program was reviewed by J. C. Haire of the SPERT Project in the March 1961 issue of *Nuclear Safety*,[73] insofar as it concerns self-shutdown mechanisms in reactors. An additional study of interest was carried out with the BSR-II reactor, where not only self-limiting excursions were observed but the effects of a fast control system were also studied to determine its effectiveness in reducing the peak power of excursions of a given initial stable period.

Following a series of static measurements in the SPERT-I facility to determine flux shape, temperature and void coefficients, and inhour measurements of the control rods, a series of transient tests was undertaken,[60,74,75] all from a subcooled zero-power initial condition. In one series of tests, the power excursion was self-limiting by the action of the inherent reactivity-compensating mechanisms of the reactor. The peak power versus reciprocal period curve for this test series was almost identical with that for a previous test series using another stainless-steel core and was in substantial agreement with the predictions of the energy model of self-shutdown.[76] In another series of tests, the control system (described by Tallackson et al.[77]) was used to limit the excursions. Two types of safety circuits were used: period scram and level scram. In order to test both, some runs used only the level scram circuits. In all cases where both were active, the period scram circuit shut the reactor down. The main conclusions to be drawn are the following:

1. For reciprocal periods up to 200 sec^{-1}, the peak power of excursions with safety circuits operating was more than five decades less than in self-limiting excursions of the same reciprocal period; with only the level scrams in operation, the peak power is reduced by more than three decades up to 100 sec^{-1}. Despite the fact that this safety system has a fast response (only about 2-msec magnet-release time, and 8-g initial control-plate acceleration), which removed 1 dollar of reactivity in the first 50 msec after scram signal, there exists a reciprocal period beyond which the control system cannot limit the excursion to peak powers any lower than the inherent shutdown mechanism provides. For the period scram, this failure threshold is at about 400 sec^{-1}, and, for the level scram, at about 250 sec^{-1}. These reciprocal periods correspond to step-reactivity transients of 2.2 and 1.75 dollars, respectively, and are associated with peak powers of about 2×10^3 Mw and 1×10^3 Mw, respectively.

2. The threshold for core damage occurs at a peak power of about 300 Mw, and a "catastrophe threshold," i.e., incipient fuel dispersion, is expected to lie at about 2000 Mw. It must therefore be concluded that, although this fast safety system can protect the reactor against damage from relatively mild excursions that would merely damage the reactor, it cannot protect the reactor against a catastrophic excursion such as would result from a step-reactivity insertion of 2.4 dollars or more.

3. In order to permit full reactivity insertion before the transient occurred, the reactor was sourceless and consisted of clean fuel. Under these conditions extremely long and variable waiting times occurred between the end of the reactivity insertion and the occurrence of the transient. In some cases involving periods of the order of 5 msec, time delays of the order of 400 to 800 msec occurred. This means that in such circumstances even relatively very slow reactivity-insertion mechanisms would suffice to cause very severe excursions since they could continue for very long times before the system responds to the supercriticality. This result focuses attention anew on the importance of having sources present in the reactor at all times, not only to give instrument readings but, more important, to bring the self-shutdown mechanisms into play before slow reactivity-insertion processes have had time to introduce large excess reactivities.

(E. G. Silver)

References

1. Fission-Product Release from UO_2, *Nuclear Safety*, 1(2): 14–15 (December 1959).
2. J. D. Eichenberg et al., Effects of Irradiation on Bulk UO_2, USAEC Report WAPD-183, Westinghouse Electric Corp., Bettis Atomic Power Laboratory, October 1957.
3. J. A. L. Robertson et al., Behavior of Uranium Oxide as a Reactor Fuel, *Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1958*, Vol. 6, p. 670, United Nations, New York, 1958.
4. J. R. MacEwan and V. B. Lawson, Thermal Simulation Experiments with a UO_2 Fuel Rod Assembly, Canadian Report CRFD-915, March 1960.
5. H. W. Newkirk and J. L. Bates, Photographic Observations of Thermal Effects in Unirradiated Uranium Dioxide Fuel Cores, USAEC Report HW-60828, Hanford Atomic Products Operation, June 22, 1959.
6. J. A. L. Robertson et al., Irradiation Behavior of UO_2 Fuel Elements, *Nuclear Met.*, 6: 45–70 (1959).
7. R. Scott, Thermal Conductivity of UO_2, British Report AERE-M/R-2526, March 1958.
8. J. A. L. Robertson, Some Consequences of Excess Oxygen in UO_2, paper presented at Symposium on Radiation Effects in Refractory Materials, 64th Annual Meeting of ASTM, Atlantic City, N. J., June 27–28, 1961.
9. J. L. Bates, Thermal Conductivity of UO_2 Improves at High Temperatures, *Nucleonics*, 19(6): 83 (1961).
10. J. T. Lamartine, ORNL, private communication to J. L. Scott, July 20, 1961.
11. A. H. Booth and G. T. Rymer, Determination of the Diffusion Constant of Fission Xenon in UO_2 Crystals and Sintered Compacts, Canadian Report CRDC-720, August 1958.
12. A. B. Auskern, The Diffusion of Krypton-85 from Uranium Dioxide, USAEC Report WAPD-TM-185, Westinghouse Electric Corp., Bettis Atomic Power Laboratory, February 1960.
13. A. B. Auskern, Further Work on the Diffusion of Kr^{85} from UO_2 Powder, USAEC Report WAPD-TM-225, Westinghouse Electric Corp., Bettis Atomic Power Laboratory, August 1960.
14. D. F. Toner and J. L. Scott, Study of the Factors Controlling the Release of Xe^{133} from Bulk UO_2, Proceedings of the Symposium on Radiation Effects on Refractory Fuel Compounds, June 27–28, 1961, American Society for Testing Materials.
15. W. H. Stevens et al., The Diffusion Behavior of Fission Xenon in Uranium Dioxide, in Nuclear Reactor Chemistry, First Conference, Gatlinburg, Tenn., Oct. 12–14, 1960, USAEC Report TID-7610, pp. 7–22, Oak Ridge National Laboratory, July 1961.
16. G. Long et al., Diffusion of Fission Products in Uranium Dioxide and Uranium Monocarbide, in Nuclear Reactor Chemistry, First Conference, Gatlinburg, Tenn., Oct. 12–14, 1960, USAEC Report TID-7610, p. 1, Oak Ridge National Laboratory, July 1961.
17. B. Lustman, Release of Fission Gases from UO_2, USAEC Report WAPD-173, Westinghouse Electric Corp., Bettis Atomic Power Laboratory, March 1957.
18. V. R. Lindner and H. J. Matzke, Some Tests on the Diffusion of Xe^{133} from Oxides of Uranium, *Z. Naturforsch.*, 13a: 794 (1958).
19. J. B. Melehan and F. A. Rough, In-Pile Fission-Gas Release from UO_2, Proceedings of the Symposium on Radiation Effects in Refractory Fuel Compounds, June 27–28, 1961, American Society for Testing Materials.
20. R. M. Carroll, The Continuous Release of Fission Gas from UO_2 During Irradiation, Proceed-

ings of the Symposium on Radiation Effects in Refractory Fuel Compounds, June 27–28, 1961, American Society for Testing Materials.

21. W. B. Cottrell et al., Fission-Product Release from UO_2, USAEC Report ORNL-2935, Oak Ridge National Laboratory, Sept. 13, 1960.
22. F. J. Stubbs and C. B. Webster, Emission of Fission-Product Gases from UO_2 During Irradiation in a Reactor, paper presented by F. J. Stubbs, AERE, Harwell, at the Libby-Cockcroft Meeting on Irradiation Effects in UO_2, Hanford Laboratory, Richland, Wash., Oct. 22–25, 1960.
23. R. M. Carroll and C. D. Baumann, Experiment on Continuous Release of Fission Gas During Irradiation (An Interim Report), USAEC Report ORNL-3050, Oak Ridge National Laboratory, Feb. 23, 1961.
24. W. B. Lewis, The Return of Escaped Fission Product Gases to UO_2, Canadian Report DM-58 (AECL-964), Jan. 21, 1960.
25. M. L. Bleiberg et al., Effects of Burnup on Certain Ceramic Fuel Materials, Proceedings of the Symposium on Radiation Effects on Refractory Fuel Compounds, June 27–28, 1961, American Society for Testing Materials.
26. J. L. Scott, The Release of Kr^{85} from UO_2 in ORR Capsules, USAEC Report ORNL-3195, Oak Ridge National Laboratory, Oct. 6, 1961.
27. J. A. L. Robertson, Understanding UO_2 Fuel Elements, *Atompraxis*, 7(4): 121–126 (1961).
28. Activity Build-Up, *Nuclear Safety*, 1(1): 32–33 (September 1959).
29. C. A. Bergmann et al., SM-1 Research and Development Program, Long-Lived Induced Activity Buildup During SM-1 Core I Lifetime, Task XVIII, Phase I, USAEC Report APAE-77, Alco Products, Inc., Nov. 30, 1960.
30. R. D. Oldenkamp et al., Evaluation of Crud Deposits on PWR Blanket Bundles, USAEC Report WAPD-TM-261, Westinghouse Electric Corp., Bettis Atomic Power Laboratory, January 1961.
31. G. P. Dix et al., The Origin and Disposal of Power Reactor Wastes, USAEC Report MND-1235, p. 39, Martin Co., Nuclear Div., Feb. 5, 1958.
32. H. Feinroth and T. D. Sutter, Jr., The First Refueling of the Shippingport Atomic Power Station, USAEC Report WAPD-233, Westinghouse Electric Corp., Bettis Atomic Power Laboratory, July 1960.
33. W. S. Brown and C. A. Bergmann, Reducing Radioactivity Buildup in PWR's, *Nucleonics*, 18(7): 60–63 (July 1960).
34. S. Yerazunis et al., Mechanisms of Reactor System Activation, USAEC Report KAPL-M-SMS-98, Knolls Atomic Power Laboratory, May 27, 1959.
35. G. P. Pancer and J. L. Zegger, Evaluation of Low Impurity Core Material in the SM-2, USAEC Report APAE-Memo-214, Alco Products, Inc., Aug. 14, 1959.
36. C. R. Bergen, Activity Buildup Control by High *p*H Coolant. A Literature Survey, USAEC Report APAE-Memo-280, Alco Products, Inc., Jan. 23, 1961.
37. C. R. Bergen and J. Chupak, Corrosion and Activity Buildup in a Power Reactor, *Ind. Eng. Chem.*, 52: 699 (July 1960).
38. G. P. Dix et al., The Origin and Disposal of Power Reactor Wastes, USAEC Report MND-1235, Martin Co., Nuclear Div., Feb. 5, 1958.
39. Stone and Webster Engineering Corp. and Combustion Engineering, Inc., Advanced Pressurized Water Reactor Study, USAEC Report TID-8502 (Pt. 1), July 1959.
40. E. E. Gross and J. H. Marable, Static and Dynamic Multiplication Factors and Their Relation to the Inhour Equation, *Nuclear Sci. and Eng.*, 7: 281–291 (1960).
41. A. B. Reynolds et al., Reactivity Effects of Large Voids in the Reflector of a Light-Water-Moderated and -Reflected Reactor, *Nuclear Sci. and Eng.*, 7: 1–13 (1960).
42. P. Braffort, The Dynamic Problems of Power Reactors and Analogue Computers, *Proceedings of the International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955*, Vol. 5, pp. 369–376, United Nations, New York, 1956.
43. G. S. Stubbs, Design and Use of Reactivity Computer, *IRE Trans. Nuclear Sci.*, NS-4 (1): 40–48 (1957).
44. C. A. Sastre, The Measurement of Reactivity, *Nuclear Sci. and Eng.*, 8: 443–447 (1960).
45. W. S. Hogan, Negative-Reactivity Measurements, *Nuclear Sci. and Eng.*, 8: 518–522 (1960).
46. A. F. Henry, The Application of Reactor Kinetics to the Analysis of Experiments, *Nuclear Sci. and Eng.*, 3: 52–70 (1958).
47. B. E. Simmons, A Survey Experiment in Reactor Pulsing Techniques, USAEC Report KAPL-M-BES-2, Knolls Atomic Power Laboratory, Jan. 9, 1957.
48. B. E. Simmons, Theory of Pulsed-Neutron Reactivity Measurements in Heterogeneous Reactors, USAEC Report KAPL-M-BES-3, Knolls Atomic Power Laboratory, July 8, 1957.
49. B. E. Simmons and J. S. King, A Pulsed-Neutron Technique for Reactivity Determination, *Nuclear Sci. and Eng.*, 3: 595–608 (1958).
50. B. E. Simmons, The Dynamic Reactivity Interpretation of Pulsed Neutron Measurements, *Nuclear Sci. and Eng.*, 5: 254–256 (1959).
51. J. Lewin, The Use of the Generation Time in Reactor Kinetics, *Nuclear Sci. and Eng.*, 7: 122–126 (1960).
52. R. Perez-Belles et al., Pulsed-Neutron-Source Measurements at the Bulk Shielding Facility, in Neutron Physics Division Annual Progress Report for Period Ending September 1, 1960, USAEC Report ORNL-3016, pp. 12–17, Oak Ridge National Laboratory, Dec. 13, 1960.

53. G. deSaussure et al., Pulsed-Neutron-Source Measurements at the TSR-II, in Neutron Physics Division Annual Progress Report for Period Ending September 1, 1960, USAEC Report ORNL-3016, pp. 54–55, Oak Ridge National Laboratory, Dec. 13, 1960.
54. G. deSaussure et al., Reactivity Worth of the Central Fuel Element in the Bulk Shielding Reactor-I, *Nuclear Sci. and Eng.*, 9: 291–298 (1961).
55. A. H. Spano, Perturbation Measurement of l/β_{eff} Using a $1/v$ Absorber, USAEC Report IDO-16638, Phillips Petroleum Co., Mar. 3, 1961.
56. E. G. Silver and J. Lewin, Safeguard Report for a Stainless Steel Research Reactor for the BSF (BSR-II), USAEC Report ORNL-2470, Oak Ridge National Laboratory, Aug. 11, 1958.
57. G. R. Keepin et al., Delayed Neutrons from Fissionable Isotopes of Uranium, Plutonium, and Thorium, *Phys. Rev.*, 107: 1044 (1957).
58. C. E. Cohn, Determination of Reactor Kinetic Parameters by Pile Noise Analysis, *Nuclear Sci. and Eng.*, 5: 331–335 (1959).
59. C. E. Cohn, A Simplified Theory of Pile Noise, *Nuclear Sci. and Eng.*, 7: 472–475 (1960).
60. F. Schroeder (Ed.), SPERT Project Quarterly Technical Report for April through June 1960, USAEC Report IDO-16640, Phillips Petroleum Co., Apr. 7, 1961.
61. C. W. Griffin and J. G. Lundholm, Jr., Measurement of the SRE and KEWB Prompt Neutron Lifetime Using Random Noise and Reactor Oscillation Techniques, USAEC Report NAA-SR-3765, Atomics International, Oct. 15, 1959.
62. J. G. Lundholm, Jr., et al., Measurement of Zero Power Frequency Response of the SRE, USAEC Report NAA-SR-3762, Atomics International, Nov. 1, 1960.
63. A. Wasserman and A. H. Spano, Pile Oscillator Design Studies, in USAEC Report IDO-16489, pp. 30–55, Phillips Petroleum Co., Jan. 19, 1959.
64. G. G. Bilodeau et al., PDQ—An IBM-704 Code To Solve the Two-Dimensional Few-Group Neutron-Diffusion Equations, USAEC Report WAPD-TM-70, Westinghouse Electric Corp., Bettis Atomic Power Laboratory, August 1957.
65. H. G. Gelbard, APCOI, Report CPM-M-135, 1958.
66. G. vonDardel and N. G. Sjöstrand, Diffusion Measurements with Pulsed Neutron Sources, in *Progress in Nuclear Energy, Series I: Physics and Mathematics*, Vol. 2, pp. 183–219, Pergamon Press, New York, 1958.
67. K. H. Beckurts, Reactor Physics Research with Pulsed Neutron Sources, *Nuclear Instr. & Methods*, 2: 144–168 (1961).
68. T. Nilsson and N. G. Sjöstrand, The Effect of a Diagonal Control Rod in a Cylindrical Reactor, *Nuclear Sci. and Eng.*, 8: 12–13 (1958).
69. R. Perez-Belles et al., An Application of the Pulsed-Neutron Technique to Reactor Study, in Neutron Physics Division Annual Progress Report for Period Ending September 1, 1960, USAEC Report ORNL-3016, pp. 136–141, Oak Ridge National Laboratory, Dec. 13, 1960.
70. T. F. Ruane et al., A Measurement of the Effective Delayed Neutron Fraction by the Substitution Method, *Trans. Am. Nuclear Soc.*, 1(2): 142–143 (1958).
71. S. Kaplan and H. F. Henry, An Experiment To Measure Effective Delayed Neutron Fractions, USAEC Report WAPD-TM-209, Westinghouse Electric Corp., Bettis Atomic Power Laboratory, February 1960.
72. R. Perez-Belles et al., The Measurement of the Effective Delayed-Neutron Fraction for the BSR-I, in Neutron Physics Division Annual Progress Report for Period Ending September 1, 1960, USAEC Report ORNL-3016, pp. 5–12, Oak Ridge National Laboratory, Dec. 13, 1960.
73. J. C. Haire, A Summary Description of the SPERT Experimental Program, *Nuclear Safety*, 2(3): 15–22 (March 1961).
74. E. G. Silver and K. M. Henry, Jr., Bulk Shielding Reactor II: Status Report, Including SPERT Tests, in Neutron Physics Division Annual Progress Report for Period Ending September 1, 1960, USAEC Report ORNL-3016, pp. 29–41, Oak Ridge National Laboratory, Dec. 13, 1960.
75. F. L. Bentzen, The Merits of Inherent Shutdown Versus Mechanical Shutdown of a Plate Type, Water-Moderated and -Reflected Reactor in a Runaway Condition, *Trans. Am. Nuclear Soc.*, 3(2): 429–430 (1960).
76. S. G. Forbes, The Dependence of Reactor Behavior on the Self-Shutdown Mode, USAEC Report IDO-16635, Phillips Petroleum Co., Apr. 25, 1961.
77. J. R. Tallackson et al., Performance Tests of ORNL Fast Safety System for Pool Type Reactors, in USAEC Report ORNL-3001, pp. 102–111, Oak Ridge National Laboratory, July 1960.

Section III

CONTROL AND INSTRUMENTATION

The Karlsruhe Symposium on Criticality Control

A symposium on Criticality Control in Chemical and Metallurgical Plants was sponsored by the European Community of Atomic Energy (Euratom) for the Organization for European Economic Cooperation (OEEC) and was held in collaboration with the Kernreaktor Bau- und Betriebsgesellschaft mbh (Nuclear Reactor Construction and Operation Co.). Meetings were held May 2–5, 1961, at the Nuclear Reactor Station in Karlsruhe, Germany.

This was the first international meeting devoted entirely to this topic, and its stature was far greater than that of any like discussions in the United States. Attendance at the symposium was limited to approximately 100 persons representing the western European countries, the United Kingdom, the United States, and international organizations with nuclear interests. The participants were primarily criticality specialists, but engineers having responsibilities for safety in operations with fissile materials were among those present.

Twenty-one papers were presented at the symposium: six from the United Kingdom, five from France, one from the International Atomic Energy Agency (IAEA), and nine from the United States. The proceedings of the symposium, including the discussion of each paper, are to be published by the OEEC in the fall, and the resumes below refer primarily to the written papers rather than to the abbreviated versions which were presented verbally at the meetings. Werner Heisenberg, Director of the Max Planck Institute for Physics and Astrophysics and Honorary Chairman of the symposium, gave the opening address, followed by Leandre Nicolaidis of Euratom.

Theoretical and Calculational Methods

A comprehensive introduction to the subject of nuclear safety, including both the physics of criticality and its application to practical problems, was presented by E. R. Woodcock (U. K.—Risley). The following paper, by R. T. Ackroyd (U. K.—Risley) and E. D. Pendlebury (U. K.—Aldermaston), surveyed in relatively detailed fashion the theoretical calculation methods employed to determine criticality. This review gave the mathematical bases for diffusion theory and its variational and perturbation techniques and for the spherical harmonics, S_n, and Monte Carlo methods. Detailed applications of these techniques were not discussed, but adequate literature references to their use were made. The expositions of each method were sufficiently complete to indicate clearly the characteristics of each. Further, the authors commented on the conditions under which each calculation can be properly applied and made recommendations regarding reducing the cost of extensive calculational programs by the use of relatively simple computational techniques taken in conjunction with the more advanced methods. The paper concluded with a note on cross-section data requirements.

A paper by P. Benoist, R. Caizergnes, Clouet d'Orval, and B. Duchemin (France—Saclay) furthered the discussion of theoretical methods. These authors placed particular stress on data requirements, the availability and characteristics of various computer programs, and the application of these calculational methods to specific types of critical systems.

Experimental Results

A review of criticality data was presented by A. D. Callihan (U. S.—ORNL). Emphasis was placed largely on experimental data obtained

recently,[1-6] particularly data from interaction tests and experiments for determining critical parameters at low U^{235} isotopic enrichments. Of particular interest was an extensive presentation of data relating critical mass, volume, and thickness to moderation and density. Many data had been assembled and correlated covering a wide range of isotopic compositions, materials, and degrees of heterogeneity. In addition, the survey included results from experiments on U^{233} and plutonium, as well as from a variety of rather specialized experiments involving U^{235} of various geometries and a number of different moderators.

Various correlations between experimental and theoretical critical data were presented by H. C. Paxton (U. S.—LASL). Among the topics discussed were the adjustment of empirical data using relatively simple theoretical models and the use of computation as a substitute for experiments. Many comparisons of the results of experiments with computed results were given, and extensive computational studies by Mills[7] and Roach[8] were reported. A comprehensive survey of the minimum critical dimensions of aqueous solutions of U^{235}, U^{233}, and Pu^{239}, based on S_n calculations, was presented. Paxton concluded with a discussion of the reliability of the computed results and the selection of safety factors when such data are to be used for designing nuclear systems and operating procedures.

A session of the symposium dealt with the facilities for experimental criticality research. The equipment, methods, organization, and types of experiments under way were discussed by Clouet d'Orval (France—Saclay), by J. G. Walford (U. K.—Dounreay) and A. F. Thomas (U. K.—Aldermaston), and by A. D. Callihan (U. S.—ORNL).

Criticality Control in Plant Design

The application of criticality control principles to plant design was discussed by J. N. Hill and H. F. Parker (U. K.—Risley), who presented the concepts involved in safe-by-shape controls and the determination of safe geometric parameters from criticality data. Examples of process and mechanical design problems which may arise in such treatment were considered, and practical applications of the safe-by-shape approach at the chemical separations plant of the Dounreay fast reactor were discussed.

Concentration control as a means of effecting nuclear safety was discussed by J. Gunther (France—Dijon). Considerable attention was given to the selection of safe concentration limits by using suitably conservative cross sections for calculation. The relation between the process chemistry and criticality problems was discussed in connection with equipment such as extraction columns in which fissionable material may appear in more than one chemical phase.

Mass batch control and its applications were described by A. J. Mallett (U. S.—UCNC). The safety factors for such mass batches, the hazards of batch operation, and the problems of sampling analysis and administrative controls were discussed. In addition, the use of mass batch controls in combination with volume and moderation controls was considered, and examples were quoted.

The somewhat controversial question of the use of nuclear poisons for nuclear safety controls in an industrial milieu was discussed in two papers. Potential homogeneous and heterogeneous poisoning methods for controlling criticality were reviewed by J. W. Wachter (U. S.—UCNC), who cited various experiments performed to evaluate poisons for nuclear safety purposes. The use of poisons as additional, primary, or contingent controls was discussed, distinctions among the various types of control being drawn according to the degree of reliance placed on the presence of the poison. A specific example of the use of fixed nuclear poisons was provided by R. Caizergues (France—Saclay), who described a theoretical study of a vessel poisoned by radial fins containing boron or cadmium. From these papers it appeared that, where the use of poisons has been thoroughly established as being mechanically, chemically, and nuclearly safe, poisoned equipment can be considered equivalent in safety to the corresponding geometrically safe vessel. To establish this safety may require a costly experimental or theoretical study, but the potential economic savings may be very great.

Specific applications of the various control principles were discussed in papers by J. W. Wachter and W. T. Mee (U. S.—UCNC) dealing with a typical plant for manufacturing enriched UO_2 fuel elements and by E. D. Clayton (U. S.—GE) concerning nuclear safety in the chemical and metallurgical processing of plutonium. In both papers the flow of fissionable material was traced through its various processing steps,

Table III-1 CRITICALITY ACCIDENTS*

Date	Location	Active material	Geometry	Total fissions	Cause	Physical damage
			Solution Systems			
December 1949	Los Alamos Scientific Laboratory, N. Mex.	~1 kg of U^{235} in $UO_2(NO_3)_2$ in 13.6 liters of water	Sphere, graphite reflected	$3-4 \times 10^{16}$	Manual withdrawal of two poison control rods	None
Nov. 16, 1951	Hanford Works, Richland, Wash.	1.15 kg of Pu in $PuO_2(NO_3)_2$ in 63.8 liters of water	Sphere, 93% full, unreflected	8×10^{16}	Poison control rod run out too fast	None
May 26, 1954	Oak Ridge National Laboratory, Tenn.	18.3 kg of U^{235} in UO_2F_2 in 55.4 liters of water	Cylindrical annulus, unreflected	1×10^{17}	Tilting of inner poison cylinder	None
Feb. 1, 1956	Oak Ridge National Laboratory, Tenn.	27.7 kg of U^{235} in UO_2F_2 in 58.9 liters of water	Cylinder, unreflected	1.6×10^{17}	Falling scram set up waves creating a critical geometry	Warping of bottom of cylinder
June 16, 1958	Y-12 Processing Plant, Oak Ridge, Tenn.	2.5 kg of U^{235} in $UO_2(NO_3)_2$ in 56 liters of water	Cylinder, concrete reflected below	1.3×10^{18}	Wash water added to $UO_2(NO_3)_2$ solution	None
Dec. 30, 1958	Los Alamos Scientific Laboratory, N. Mex., Pu Processing Plant	3.27 kg of Pu in $PuO_2(NO_3)_2$ in ~168 liters of water	Cylinder, water reflected below	1.5×10^{17}	Agitator created a critical geometry	None
Oct. 16, 1959	Chemical Processing Plant, Idaho Reactor Testing Area	34.5 kg of U^{235} in ~800 liters of $UO_2(NO_3)_2$ and water	Cylinder, concrete reflected below	$\sim 4 \times 10^{19}$	Solution siphoned from safe to unsafe geometry	None
Jan. 25, 1961	Chemical Processing Plant, Idaho Reactor Testing Area	8 kg of U^{235} in $UO_2(NO_3)_2$ in 40 liters of water	Cylinder	6×10^{17}	Solution pumped from safe to unsafe geometry	None
			Metal Systems in Air			
Aug. 21, 1945	Los Alamos Scientific Laboratory, N. Mex.	6.2 kg of δ-phase Pu	Spherical core, tungsten-carbide reflected	$\sim 10^{16}$	Hand stacking of reflector	None
May 21, 1946	Los Alamos Scientific Laboratory, N. Mex.	6.2 kg of δ-phase Pu	Spherical core, beryllium reflected	$\sim 3 \times 10^{15}$	Hand stacking of reflector	None
Apr. 18, 1952	Los Alamos Scientific Laboratory, N. Mex.	92.4 kg of uranium metal enriched to 93% U^{235}	Cylinder, unreflected	1.5×10^{16}	Computational error	None
Feb. 3, 1954	Los Alamos Scientific Laboratory, N. Mex.	53 kg of uranium metal enriched to 93% U^{235}	Sphere, unreflected	5.6×10^{16}	Incorrect operation	Slight warping of pieces
Feb. 12, 1957	Los Alamos Scientific Laboratory, N. Mex.	54 kg of uranium metal enriched to 93% U^{235}	Sphere, unreflected except for experiment	1.2×10^{17}	Shift of experiment	Warping, oxidation, near melting close to center
			Inhomogeneous Water-Moderated Systems			
June 4,						

						ing close to center

Inhomogeneous Water-Moderated Systems

June 4, 1945	Los Alamos Scientific Laboratory, N. Mex.	35.4 kg of uranium enriched to ~83% U^{235}, ½-in. cubes	Pseudosphere, water reflected	$\sim 3 \times 10^{16}$	Water seeping between blocks	None
Feb. 1, 1951	Los Alamos Scientific Laboratory, N. Mex.	2 cylinders of uranium, 24.4 and 38.5 kg enriched to 93% U^{235}	2 cylinders, water reflected	10^{17}	Scram increased reactivity	Slight oxidation
July 6, 1952	Argonne National Laboratory, Argonne, Ill.	6.8 kg of U^{235} in oxide particles in plastic	Inhomogeneous cylinder, water reflected	1.22×10^{17}	Manual withdrawal of central safety rod	Plastic destroyed
Dec. 12, 1952	Chalk River, Canada	Normal uranium	Rods, D_2O moderated, graphite reflected	1.2×10^{20}	Safety circuits failed, control rod misoperated	Core ruined
July 22, 1954	Reactor Testing Area, Idaho	U-Al plates clad with Al	Inhomogeneous cylinder, water moderated	4.68×10^{18}	Estimate of expected excursion too low	Reactor destroyed
Oct. 15, 1958	Vinca, Yugoslavia	3996 kg of normal uranium	Rods, D_2O moderated, unreflected	2.5×10^{18}	Uncontrolled pumping of D_2O	None
Jan. 3, 1961	Reactor Testing Area, Idaho	U-Al plates clad with Al	Inhomogeneous cylinder, water moderated	1.5×10^{18}	Not yet determined	Not yet determined

Miscellaneous Systems

Feb. 11, 1945	Los Alamos Scientific Laboratory, N. Mex.	UH_3 pressed in Styrex	Cylinder	$\sim 6 \times 10^{15}$	Reflector added or source too large or both	UH_3-Styrex cubes swollen and blistered
1953	U.S.S.R.	Unknown	Unknown	Unknown	Unknown	Unknown
Nov. 29, 1955	Reactor Testing Area, Idaho	½-in. U^{235} rods	Cylinder, rods cooled by NaK	4.7×10^{17}	Incorrect scram used	Core molten
July 3, 1956	Los Alamos Scientific Laboratory, N. Mex.	58 kg of uranium enriched to 93% U^{235}, 2- and 5-mil foils	Cylinder	3.2×10^{16}	Too rapid assembly	None

*W. R. Stratton, A Review of Criticality Accidents, in Proceedings of the Symposium on Criticality Control in Chemical and Metallurgical Plants, Karlsruhe, Germany, May 2–5, 1961, to be published.

Table III-2 ANALYSIS OF CRITICALITY ACCIDENTS LISTED IN TABLE III-1*

Incident	Observed fission yield	Spike yield, fissions	Plateau yield, fissions	Max. α, sec^{-1}	Time interval from prompt critical to peak power, sec	Width of spike at one-half max. value, sec	Time duration of plateau, sec	Peak power during spike, fissions/sec	Initial plateau power, fissions/sec	Probable quenching mechanism
Solution Systems										
Water-boiler reactor control rods removed by hand, December 1949	$3-4 \times 10^{16}$	$\sim 3 \times 10^{15}$	$2.7-3.7 \times 10^{16}$	6.25	~3.5		~1.5		$2-3 \times 10^{16}$	Rise of neutron temperature
Plutonium solution assembly, cadmium rod removed too rapidly, Nov. 16, 1951	8×10^{16}	8×10^{16}	0.0	No estimate	No estimate	No estimate	0.0	No estimate	0.0	Microbubbles, neutron temperature, and poison rod
Uranium solution assembly, central poison cylinder tilted to less favorable position, May 26, 1954	10^{17}	5×10^{16}	5×10^{16}	4.8×10^{2}	7.8×10^{-2}	5.4×10^{-3}	0.5	8.2×10^{18}	4.8×10^{16} to 1.4×10^{17}	Microbubbles and neutron temperature
Uranium solution assembly, wave motion created by falling cadmium sheet, Feb. 1, 1956	1.6×10^{17}	1.6×10^{17}	0.0	6.1×10^{3}	7.4×10^{-3}	3.7×10^{-4}	0.0	$\sim 3.5 \times 10^{20}$	0.0	Microbubbles and neutron temperature
Uranium process solution combined with wash water in 55-gal drum, June 16, 1958	1.3×10^{18}	$\sim 10^{16}$	1.29×10^{18}	No estimate	No estimate	No estimate	170.0	No estimate	No estimate	Microbubbles, neutron temperature, and boiling
Separated phases in plutonium process tank, Dec. 30, 1958	1.5×10^{17}	1.5×10^{17}	0.0	No estimate	No estimate	No estimate	0.0	No estimate	0.0	Microbubbles, neutron temperature, and stirring action
Uranium process solution siphoned from safe to unsafe geometry, Oct. 16, 1959	4×10^{19}	$\sim 10^{17}$	$\sim 4 \times 10^{19}$	No estimate	No estimate	No estimate	No estimate	No estimate	No estimate	Microbubbles and neutron temperature
Uranium process solution pumped from safe to unsafe geometry, Jan. 25, 1961	6×10^{17}	No estimate	No estimate	No estimate	No estimate	No estimate	No estimate	No estimate	No estimate	Microbubbles and neutron temperature
Metal Systems in Air										
Plutonium core reflected with tungsten carbide, Aug. 21, 1945	10^{16}	$\sim 1.8 \times 10^{15}$	$\sim 8.2 \times 10^{15}$	$\sim 10^{4}$	$\sim 2 \times 10^{-3}$	$\sim 3.5 \times 10^{-4}$	0.8	$\sim 5 \times 10^{18}$	$\sim 1.2 \times 10^{16}$	Thermal expansion
Plutonium core reflected with beryllium, May 21, 1946	3×10^{15}	$\sim 1.8 \times 10^{14}$	$\sim 2.86 \times 10^{15}$	$\sim 10^{3}$	1.5×10^{-2}	$\sim 3.5 \times 10^{-3}$	0.28	$\sim 5 \times 10^{16}$	$\sim 1.2 \times 10^{16}$	Thermal expansion
Jemima, a cylindrical unreflected U^{235} assembly,	1.5×10^{16}	$\sim 10^{15}$	$\sim 1.4 \times 10^{16}$	8×10^{3}	4.3×10^{-3}	4.4×10^{-4}	≤ 0.3	2.3×10^{18}	$\leq 10^{17}$	Thermal expansion

Plutonium core reflected with beryllium, May 21, 1946	3×10^{18}	$\sim 1.8 \times 10^{18}$	$\sim 2.86 \times 10^{18}$	$\sim 10^{3}$	1.3×10^{-3}	9.25×10^{-5}	0.25	[illegible]	[illegible]	[illegible]
Jemima, a cylindrical unreflected U^{235} assembly, Apr. 18, 1952	1.5×10^{16}	$\sim 10^{16}$	$\sim 1.4 \times 10^{16}$	8×10^{3}	4.3×10^{-3}	4.4×10^{-4}	≤ 0.3	2.3×10^{18}	$\leq 10^{17}$	Thermal expansion
Godiva, a bare U^{235} sphere, Feb. 3, 1954	5.6×10^{16}	5.6×10^{16}	0.0	1.6×10^{5}	2.4×10^{-4}	1.8×10^{-5}	0.0	3.5×10^{21}	0.0	Thermal expansion
Godiva, a bare U^{235} sphere, Feb. 12, 1957	1.2×10^{17}	1.2×10^{17}	0.0	2.1×10^{5}	1.9×10^{-4}	1.2×10^{-5}	0.0	10^{22}	0.0	Thermal expansion
Inhomogeneous Metal Systems in Water										
Psuedosphere of uranium cubes, water reflected, June 6, 1945	$3-4 \times 10^{16}$	$\sim 3 \times 10^{15}$	$2.7-3.7 \times 10^{16}$	No estimate	No estimate	No estimate	~5.0	No estimate	No estimate	Boiling
Critical separation experiment, two large U^{235} masses in water, Feb. 1, 1951	10^{17}	$\sim 6 \times 10^{15}$	$\sim 9.5 \times 10^{16}$	No estimate	No estimate	No estimate	~0.1	No estimate	$\sim 10^{17}$	Boiling
UO_2 particles in plastic, water moderated, July 6, 1952	1.22×10^{17}	1.22×10^{17}	0.0	100	~0.6	1.9×10^{-2}	0.0	5.4×10^{18}	0.0	Plastic bubbles
NRX reactor, normal uranium in D_2O, graphite reflected, Dec. 12, 1952	1.2×10^{20}	Not prompt critical	Not prompt critical	~0.33	Not prompt critical	~30	0.0	$\sim 3 \times 10^{18}$	0.0	D_2O drained from core
BORAX reactor, Al-U alloy, water moderated, July 22, 1954	4.68×10^{18}	4.68×10^{18}	0.0	385	7.9×10^{-2}	6.5×10^{-3}	0.0	6.6×10^{20}	0.0	Boiling
Critical assembly of normal uranium rods in D_2O, unreflected, Oct. 15, 1958	2.5×10^{18}	Not prompt critical	Not prompt critical	No estimate	Not prompt critical	No estimate	0.0	No estimate	0.0	Cadmium rods inserted in core
SL-1 reactor, Al-U alloy, water moderated, Jan. 3, 1961	1.5×10^{18}	1.5×10^{18}	0.0	No estimate	No estimate	No estimate	0.0	No estimate	0.0	Boiling
Miscellaneous Systems										
The Dragon reactor, UH_3 pressed in Styrex, Feb. 11, 1945	$\sim 6 \times 10^{15}$	$\sim 6 \times 10^{15}$	0.0	No estimate	$\sim 10^{-2}$	No estimate	0.0	No estimate	0.0	Thermal expansion
Experimental reactor, U.S.S.R., 1953	Unknown	Unknown	Unknown	Unknown	Unknown	Unknown	Unknown	Unknown	Unknown	Unknown
EBR-I. fast-breeder reactor. U^{235} rods in NaK, Nov. 29, 1955	4.7×10^{17}	Not prompt critical	Not prompt critical	~3.6	Not prompt critical	~1.0	0.0	$\sim 3 \times 10^{17}$	0.0	Reflector dropped from core
The Honeycomb, U^{235} foils moderated with graphite, July 3, 1956	3.2×10^{16}	3.2×10^{16}	0.0	~40	~1.1	0.1	0.0	9.4×10^{17}	0.0	Motion of safety rods

*W. R. Stratton, A Review of Criticality Accidents, in Proceedings of the Symposium on Criticality Control in Chemical and Metallurgical Plants, Karlsruhe, Germany, May 2–5, 1961, to be published.

and the nuclear safety problems of each step were discussed at some length. Although such descriptions are highly specific, they serve to illustrate that practical nuclear safety remains something of an applied art.

The problems involved in the storage and transport of fissionable material were considered in the next session. A paper by R. T. Ackroyd (U. K.—Risley), F. R. Charlesworth (U. K.—Ministry of Power), and A. F. Thomas (U. K.—Aldermaston) dealt with methods of calculation of interacting arrays, with special reference to storage problems. The most frequently used theoretical and empirical approaches to the problem were discussed in considerable detail. The well-known solid-angle method was described, as well as two unpublished methods: the "density analogue approach" of H. C. Paxton and the "interaction rules" of D. Kenyon. Greatest emphasis was placed, however, on the unified interaction theory developed by Ackroyd et al.[9] and the interaction parameter method of Scriven and Thomas.[10] The authors concluded that empirical methods of solid angle, density, etc., suffice where plant operations require relatively open spacing of fissile material to enable loading and unloading of storage areas. Where tighter spacings are of advantage, they could be obtained through the greater accuracy of more sophisticated calculations. The latter approach should be adopted, however, when considering a plant having a wide variety of vessels of fissile materials of differing moderations. In such a case it is necessary to examine in detail the effects of such factors as spacing, geometry, and interspersed moderators, notwithstanding the relatively high cost of elaborate computation.

The problems that arise in considering such interacting arrays were brought out in a discussion of regulations to be proposed by the IAEA to govern the international transport of radioactive and fissionable material. The proposed regulations were prepared by R. F. Barker and were presented by C. Erginsoy (IAEA). Much of the discussion dealt with the proposals themselves, which have since been published,[11] but the technical bases for the proposals were also considered in some detail, and the safety factors were discussed.

The organizational structure by which criticality control is effected was described for their several countries by P. LeCorche and R. Caizergues (France—Saclay), H. Howells and J. T. Daniels (U. K.—Risley), and C. L. Schuske (U. S.—Dow). These papers dealt with the administrative aspects of such controls and, in addition, considered such general methods of control as instrumentation, inventory accountability, and batch limitations.

Criticality Accidents

A review of criticality accidents which was presented by W. R. Stratton (U. S.—LASL) was based on an earlier review.[12] Several accidents were described and discussed. Tables III-1 and III-2, taken from this paper, summarize the accidents on which the discussion was based. Conclusions were drawn regarding the characteristics of prompt excursions and the quenching mechanisms that terminate the reactions. Such practical considerations as the use of a neutron source to limit the burst intensity were also discussed.

Conclusions

The discussions that took place following each paper were of great interest. It appeared that, whereas installations in the United States have utilized the experimental approach, greater efforts have been applied elsewhere to the calculation of critical conditions, primarily because of limited fissile material. The result has been the application of correspondingly greater safety factors. Moreover, the relative cost of establishing the nuclear safety of a production plant in comparison with the over-all initial investment appears, at the present time, considerably greater in the United Kingdom than in the United States.

An important benefit of these discussions was the direct exchange of specific experimental data. The cost of both theoretical and experimental data acquisition has become so great that the opening up of channels of communication among the participants will be of continuing mutual benefit.

The proceedings of the symposium, when published, will be the most comprehensive treatment of the theoretical and practical aspects of nuclear safety that has yet appeared. As such the proceedings will be of great benefit for those entering the field of fissionable material processing and a valuable reference work for those presently engaged in nuclear safety problems.

(J. W. Wachter)

Measurement of Control-Rod Effectiveness

The potential nuclear hazard arising in the operation of chain-reacting systems can be directly related to the amount of excess reactivity available and the method of its control. The worth of a control rod in a reactor is therefore a measure of the reactivity capable of being suppressed, as well as an indication of the margin of safety in the reactor operation. The prominence of control-rod effectiveness in the evaluation of nuclear safety warrants this presentation of the current status of methods of its measurement. Reactors in which the active core is not large are amenable to most of the experimental procedures practiced today; however, with very large reactor cores, aside from calculational techniques, the absolute determination of control-rod worth has remained a "gray" area. Recent application to this problem of the PCTR-k_∞ measuring technique that is available through the use of the Physical Constants Testing Reactor at Hanford appears to be promising.

Description of Measurement Techniques

The rod-oscillation technique is an application of pile-oscillator[13,14] measurements to a reactor in which an absorbing rod, being displaced periodically, plays the role of an oscillating absorber. In practice, the associated oscillating signal from a suitable detector is recorded. The relation between changes in amplitude and reactivity increments is experimentally established by associating a change in amplitude with a known reactivity change. This method is advantageous because the time required for each measurement is short (~5 min), and the reactor power need not be constant since an alternating component of the power is being measured rather than the power itself. The small deviation from an average power level during the course of a measurement obviates the need for equilibrium of the delayed-neutron emitters. Although comparison of the amplitudes of periodic signals derived from successive measurements on several rods yield only relative values of their effectiveness, Jankowski et al.[15] have presented an absolute evaluation by solving the kinetic equations and expressing the results in terms of a Fourier series. This absolute evaluation is applicable only to oscillations of small amplitude.

One of the early standard methods used for rod-worth measurements was the rod-drop technique. It is simple and direct. The rod is rapidly inserted into a reactor operating at a constant neutron level, and the decreasing counting rate, C, is observed as a function of time. When the counting rate is extrapolated back to the time of rod insertion, its worth may be expressed as

$$\frac{\rho}{\beta} = \frac{C_1 - C_2}{C_2} \tag{1}$$

where the subscripts 1 and 2 denote the counting rate, C, before and after insertion, respectively, ρ is the reactivity, and β is the delayed-neutron fraction. It is assumed that the change in reactivity is instantaneous and that the prompt-neutron lifetime is short compared with the lifetime of the shortest delayed group. Usually, the finite time required for insertion causes uncertainty in the evaluation of C_2. Kolar and Kloverstrom[16] eliminate this difficulty by using the determined stable period, T, and the inhour equation in the form

$$\rho = \frac{l^*}{T} + \gamma \sum_1^6 \frac{\beta_i}{1 + \lambda_i T} \tag{2}$$

where γ is the ratio of effectiveness of delayed neutrons to prompt neutrons, l^* is the prompt-neutron lifetime when the system is critical, and λ_i and β_i are the decay constant and delayed-neutron fraction for each of the six delayed-neutron groups, respectively.

In the rod bump-period technique, with the reactor critical and at a constant neutron level, the rod to be evaluated is displaced a small amount and the associated reactor period is determined. The inhour equation then yields the associated reactivity, and this in turn is related to the rod insertion distance by means of first-order perturbation theory, which adequately[16] expresses the differential effectiveness relation. The major disturbing feature appears to be the time consumed, although the problem of establishing the constant neutron level between increments can be avoided, thus eliminating an elusive time variable. For example, Williams et al.,[17] by means of a relative measurement of two control rods in the APPR critical experiments, obtained a calibration from successive measure-

ments of positive and negative periods resulting from alternate compensating displacements.

The pulsed-neutron technique has been adequately described by Simmons.[18,19] Succinctly, a subcritical multiplying system is subjected to a burst of neutrons, and the resulting stable decay of the neutron population is observed. The observation begins after the initial transients of the burst have died out and continues until the delayed neutrons dominate the total neutron population. Over the time of measurement, the behavior is exponential. The resulting relation between the decay constant and the reactivity is

$$\alpha = \alpha_0 \left(1 - \frac{\rho}{\gamma\beta}\right) \qquad (3)$$

where $\alpha_0 = \gamma\beta/l^*$ is the decay constant at delayed critical.

The source-jerk technique is similar to the rod-drop technique, except that the reactor is subcritical and the constant neutron level is that due to the multiplication of the neutron source present. The source is suddenly removed, and the procedure outlined in connection with Eq. 1 is followed. The limitations of this method stem directly from source strength and its effect on detector response to a weak neutron field.

The PCTR-k_∞ measurement[20,21] has been extended to the evaluation of negative reactivities and, thus, is directly applicable to measurements of control-rod strengths. Briefly, the PCTR is a null reactivity test assembly consisting of three concentric regions: a central test cavity surrounded by a buffer zone which, in turn, is surrounded by the reactor composed of enriched-uranium rods in graphite. The procedure is to first establish the neutron flux or adjoint flux spectrum characteristic of the test material by means of the buffer region (the material to be placed in the cavity may also be used here). Then the test material is poisoned (usually with copper) until there is no observed change in reactivity with or without the test material in the cavity. When applied to control-rod evaluation, a supercell (made up of unit cells of the material under study) is poisoned until $k_\infty = 1$. Then the control rod is inserted and the necessary amount of poison is removed until no change in reactivity occurs. The rod may be of such strength that a negative amount of poison is associated with the measurement, but the results remain interpretable.[22]

Comparisons of Techniques

As a desirable step in the AEC Control-Rod Materials Program, various dynamic techniques for measuring control-rod worth were compared. Ballowe et al.[23] report a set of experiments in which four methods were chosen for intercomparison: rod drop, rod bump period, rod oscillator, and pulsed neutron. The results disclosed that the rod-drop and rod-oscillator techniques are dependent on the detector location. The exact location of the detector was not given, although it appears to have been near the reactor core and was varied along the axis parallel to the motion of the rods. The dependence was explained on the basis of a one-group diffusion-theory model which showed the signal arising from a local change in the core to satisfy a differential equation differing from that of the steady-state flux in that the signal is a function of the prompt neutrons alone. Consequently there is no reason to expect the ratio of signal to steady-state flux to be unity. The rod-bump-period and pulsed-neutron techniques do not exhibit this dependence, being, in themselves, eigenvalue determinations. The observed dependence is contrary to the earlier work of Jankowski et al.,[15] who compared the rod-drop, rod-oscillator, and source-jerk techniques. The reported results were all within experimental error, and no dependence on detector location was inferred from the independence of the constant relating power-oscillation amplitudes to reactivities. However, all errors in the rod-oscillation data were observed to be in the same direction and therefore suggestive of a systematic error.

The pulsed-neutron technique was compared with the rod-drop and rod-bump-period methods by Kolar and Kloverstrom[16] of Livermore. They confirm the work of others who have examined the technique; namely, its usefulness in evaluating large negative reactivities, as well as its speed and convenience. Important by-products of this technique, as illustrated by their work, are the quantities ρ/γ and l^*/γ. When the constants are used with the reactor kinetic equations, a calculated curve that is obtained for neutron population versus time subsequent to the rod drop aids in the evaluation of the data. For the rod-drop measurements these authors positioned the detector "about 8 ft from the reactor in locations where inverse multiplication measurements had been found to be independent of

counter location." It is possible that detector dependence did not manifest itself in the data, since comparison of the data, the calculated curve, and the calculated points, which were assumed to have varying error magnitudes, seems to indicate a reactivity resolution (±3 per cent) for the method.

The major disadvantages of these methods remain when they are applied to very large reactors. Further, the pulsed-neutron technique, which in this application becomes similar to the source-jerk method, suffers from insufficiency of neutron source strength. In comparison the PCTR-k_∞ measurements appear to be applicable to large reactors. Nichols and Heineman[24] have reported on a set of experiments in which an EGCR control rod was evaluated in the PCTR with the use of a supercell containing only 16 fuel elements and a centrally located control rod. An attractive feature of the technique was thus forcefully demonstrated, when it is considered that only ~1 per cent of the core was represented. (J. T. Thomas)

In-Core Flux Monitoring

The spatial distribution of the neutron flux in a reactor core is a valuable parameter for the evaluation of fuel-element and reactor-control-system performance. Information regarding fuel enrichment and distribution for large core designs can often be determined in research reactors and critical assemblies. Flux plots are needed for power reactors in order to optimize the fuel loading and burnup cycle through flux flattening by control-rod correction, which may also yield more uniform channel heating. Devices for ascertaining the flux distribution in a reactor core are known as in-core flux monitors. A number of different monitors currently in use or under development have been discussed in recent literature.

A standard flux-monitoring practice in research reactors is to insert activation foils or wires into the reactor core and then remove them for counting. The use of threshold detectors and associated counting techniques in the determination of thermal- and fast-neutron-flux densities and energy spectra has been discussed tutorially by Price.[25] More recently, an improved technique for the measurement of fast neutrons by foil activation was reported by Ricabarra et al.,[26] who used phosphorus instead of sulfur as the activated material. The flux required for the same activity is reduced by a factor of 70 when phosphorus is used.

Mechanical arrangements[27] for the insertion of activation wires are in standard use at Savannah River, Hanford, and Shippingport. At Yankee,[22] thimbles are provided for remotely operated devices for traversing activation wires.[27] An example of an automatic, high-resolution counter for activated ribbon has been described by Mulot.[28] A resolution as low as 0.5 mm is achieved by the use of interchangeable collimators. For most applications, that resolution would seem to be more than adequate. Researchers at Savannah River have experimented with neutron- and gamma-sensitive "thermometers," which are devices that utilize a thermocouple to measure the heat that is generated in a metallic target by the interaction of gamma rays or neutrons.[27] However, the time response of these devices is too long for the required monitoring application, their dynamic range is limited, and, in the case of the neutron thermometer, life is limited by burnup. For these reasons Savannah River prefers gamma ion chambers.

An article by Schilling[29] describes a successful device consisting of a copper-constantan thermocouple welded into a platinum pill filled with a pulverized mixture of 55 per cent aluminum and 45 per cent uranium of 20 per cent enrichment. The thermocouple junction, in series with a reference junction, is enclosed in a tiny tube at the end of a long probe handle through which the signal leads are drawn. The probe is calibrated against foils, and the thermal electromotive force is read with a galvanometer. This device has shown only 1 per cent drift in sensitivity after 300 Mw-hr of operation in the FRM reactor (Munich reactor) at a flux of approximately 1×10^{13} neutrons/(cm^2)(sec).

At Savannah River and at Dresden, work has been carried out on the use of gamma ion chambers.[27] The gamma chamber yields a signal that the authors state is essentially proportional to local power density. Flux tilts are determined by variations in the chamber signals at various points in the core. Savannah River reactors are employing, on a developmental basis, assemblies which contain three chambers and which are placed in thimbles in the reactor core. The results have been quite satisfactory. At Dresden the control system employs a matrix of 64 chambers at fixed positions in four planes

throughout the core. It is intended that the reactor control rods be programmed from these readings and that the safety system be operated from these in-core chambers. In November 1960 these chambers, which also check out the operation of the control rods, detected the breakage of a control rod.[30] It appears that this system is effective for flux gradient control and rod checking. A commercial manufacturer is marketing a chamber of this type and developing improved high-temperature cables for use with it.

These recent advances in the state of the art of in-core flux monitoring indicate only that standard techniques have been improved, while relatively little in the way of new equipment has been developed. It has been reported[27] that AEC-sponsored work is in progress on microwave attenuation from ionization and heat-transfer null-balance detectors using neutron heating, and it is hoped that some breakthroughs will be reported in the near future. (J. L. Kaufman)

Nonnuclear In-Core Instrumentation

A meeting sponsored by the AEC Division of Reactor Development was held in Washington on Apr. 28–29, 1960, to discuss in-core instrumentation for power reactors. The objectives of the meeting were (1) to review the present status of the current development programs for in-core instrumentation; (2) to discuss the major problems in this area; (3) to review present and future requirements for in-core instrumentation, particularly in power reactors; and (4) to provide a free exchange of information and ideas that might lead to improvements in reactor core performance and reactor safety through improvement of in-core instrumentation.[27] The proceedings of the meeting contain numerous references to reports that provide much additional detailed information on the various subjects. The meetings consisted of informal question and answer sessions, and the report of the meetings presents a recording of those discussions. No formal papers were presented.

This review describes the state of the art for nonnuclear instrumentation as it was discussed in the meeting. Devices for measuring flow, liquid level, differential pressure, and temperature are described, in that order.

Flow

Pitot tubes have been used in-core to measure gas flow in fuel channels of gas-cooled reactors and water flow in boiling-water reactors. Before being installed in the reactor core, the tubes are calibrated out-of-pile in an identical configuration to assure accuracy. Stauschiebe tubes, which are small cylinders placed perpendicular to the direction of flow, with pressure taps facing upstream and downstream, are also used for measurements of water flow rates; these tubes must be calibrated.

Turbine flowmeters can be used to measure the velocity of water in the inlet and in the outlet of a fuel channel with an accuracy of 1 per cent. By knowing the difference in the inlet and outlet water velocities, the steam void at the outlet of the fuel channel can be calculated. Prior to in-pile use, pairs of turbine flowmeters are calibrated with water and with water-air mixtures in an out-of-pile electrically heated loop. The steam voids calculated for void fractions ranging from 0.24 to 0.71 are within ±10 per cent of the values determined by the gamma-attenuation technique.

Turbine flowmeters have worked well in a flux of 10^{12} neutrons/(cm^2)(sec). One application was described in which, after exposure to an integrated flux of 4×10^{18} neutrons/cm^2 at 975°F while measuring the flow rate of CO_2, there was no evidence of damage to the flowmeter. The flow rates of molten salts and liquid metals have been measured with experimental turbine flowmeters for periods of more than 1000 hr at 1500°F.

Venturi tubes are commonly used to measure flow in fuel channels. As many as 42 in-core Venturi meters have been manifolded out-of-pile to one differential-pressure transmitter.

Liquid Level

Resistance type level-measuring elements have been used for measurements of fluids with high electrical conductivity. Such elements have operated satisfactorily for periods of up to 20,000 hr at 1300 to 1500°F while immersed in sodium and NaK. The accuracy of measurement over a 7-in. span of NaK and of sodium has been reported[31] to be 1 per cent.

Acoustic level-measuring devices are available for use at temperatures up to 950°F. The effect of irradiation on these devices has not been investigated.

Differential Pressure

Differential-pressure transmitters located outside the reactor vessel of a boiling-water reactor and connected to a number of pressure taps inside the vessel have been used to measure the carryover of steam, vapor holdup, downcomer velocities, and average steam voids in risers. The individual pressure tap tubes are contained in a duct. A small stream of cooling water flows over the pressure tap tubes within the duct to keep the liquid in the tubes from flashing. A large flow of cooling water cannot be used because it would cause undesirable axial temperature gradients in the duct.

Temperature

Chromel-Alumel thermocouples in stainless-steel sheaths are the thermocouples most often used as sensing elements to measure in-core temperatures in all types of reactors. The thermocouple insulation is usually magnesium oxide, although aluminum oxide insulation is often used in pressurized- and boiling-water reactors because aluminum oxide will not destructively swell if water gets into the sheath. The sheathed thermocouple assemblies used most often are $\frac{1}{16}$ in. in outside diameter, although assemblies as small as 0.040 in. in outside diameter have been used.

The guide tube, which is an extension of the reactor pressure vessel and through which thermocouples can be inserted into the core, is a significant development in core temperature measuring. Such tubes permit the replacement of thermocouples at any time during the life of the core. A $\frac{1}{16}$-in.-OD sheathed thermocouple can be inserted in a guide tube that is 25 ft long and has many turns.

The reliability of 50 sheathed Chromel-Alumel thermocouples during a period of a year of operation in a pressurized-water reactor core has been studied. The variations of the 50 individual thermocouples gave temperature values that were within 1°F of the average (95 per cent confidence limit) and within 2.5°F of an external reference temperature. The irradiation dose to these thermocouples was more than 10^{21} neutrons/cm^2. Experience with sheathed Chromel-Alumel thermocouples in liquid metal at 1500°F has also shown that they are reliable and that they hold their calibration when exposed to an irradiation dose of 10^{21} neutrons/cm^2.

If the leads from a small-diameter sheathed thermocouple, about 0.050 in. in outside diameter, pass through a zone having a temperature of 2500°F or more and the junction is in a cooler zone, the electromotive force from the thermocouple will be proportional to the hot-zone temperature. The high temperature causes the insulation resistance to become low enough to form another junction which is in parallel with the true junction.

To obtain an accurate measurement of the temperature of a fuel-element surface, the sensing element must be calibrated out-of-pile using the same materials and configuration as used in-core. Several techniques have been used to measure surface temperature in flowing gas. One technique is to flatten a 0.030-in.-OD sheathed thermocouple to 0.020 in., fasten it to the surface of the fuel element, and put a ramp upstream to give smooth flow and avoid error-causing turbulence. Another technique is to place the junction end of the sheath downstream, with the thermocouple wires extended from the sheath and spot welded to the surface of the fuel element. With this technique, a surface temperature of 1300 ± 50°F has been measured. However, in an organic coolant, the surface temperature of a fuel element has been measured to within 25°F in the 400–700°F range, based on out-of-pile calibration using 0.010-in.-OD thermocouple wires fastened to the surface.

The central temperatures of fuel plates for pressurized- and boiling-water reactors have been measured by the so-called "blocked-channel" technique. A special zirconium plate that has the thickness of two standard plates plus one water channel is fabricated into the fuel element. The water-channel space in this special plate is filled with a nonfissionable material in which the thermocouples are embedded. As many as 100 thermocouples have been spaced along a 4-ft plate. If stainless steel were used in place of zirconium, the resulting flux depression would give a temperature measurement error as great as 12 per cent.

There is a need for techniques and materials that will make it possible to measure fuel temperatures up to 4500°F in uranium oxide, uranium carbide, and other possible fuel materials. At the present time, central temperatures up to 3800°F in UO_2 pellets have been measured for a week or so and up to six weeks in the temperature range of 2500 to 3200°F. In these measurements the thermocouple was tungsten

versus rhenium insulated with beryllium oxide and contained in a swaged tantalum sheath. When the junction where the sheath and thermocouple wires were fused together was grounded, the couple often failed because of differential thermal expansion, whereas thermocouples with ungrounded junctions failed less often. The thermocouple wires entered the capsule through a hermetic electric seal so that the fission gases would be contained. Generally, the tantalum sheath showed deterioration after six weeks at temperature and exposure to fission gases. Over the six-week test period, the thermocouples indicated that the central UO_2 temperatures were decreasing. This may have been caused by changes in the pellets or in the thermocouples or both.

A pneumatic system for the measurement of gas temperatures was discussed. The system consists of two sonic nozzles in series through which a stream of gas flows. The first nozzle is located where it is desired to measure the temperature; the second is at some convenient location and is connected to the first nozzle by a tube. The pressures upstream of each nozzle and the temperature upstream of the second nozzle are measured. The desired temperature is calculated from the relation

$$T_1 = K\left(\frac{P_1}{P_2}\right)^2 T_2$$

where K is the general constant of the system. This temperature-measuring system was discussed in detail in an earlier issue of *Nuclear Safety*.[32]

There are numerous special problems and techniques associated with the determination of in-pile temperatures. Sometimes it is necessary to use odd combinations of thermocouple wires to match the thermal expansion of wires, sheath, and fuel plates. Further, platinum versus platinum-rhodium thermocouples have shown errors because of transmutation and, in high-temperature applications, because of the migration of rhodium. In another situation, resistance thermometers of platinum will be used in some pressurized-water reactors to measure fuel-channel outlet temperatures and the difference in temperature between the inlet and outlet of a channel.

A number of high-speed scanning systems will be used to monitor in-core temperatures in the near future. Plans are to use 18-channel recording oscillographs, a 100-channel oscilloscope with a mercury-jet switch which can scan 100 thermocouple stations, and an analog-to-digital conversion unit with magnetic-tape recording. Experience with current models of jet switches has shown that they must be cleaned after being in service three months because of contamination by the mercury. (H. J. Metz)

References

1. J. T. Mihalczo and J. J. Lynn, in Neutron Physics Division Annual Progress Report for Period Ending September 1, 1960, USAEC Report ORNL-3016, pp. 73–76, Oak Ridge National Laboratory, Dec. 13, 1960.
2. R. Gwin and D. W. Magnuson, Critical Experiments for Reactor Physics Studies, USAEC Report CF-60-4-12, Oak Ridge National Laboratory, Sept. 16, 1960.
3. J. C. Smith et al., Criticality of 30% Enriched Uranium Solutions in Cylindrical Geometry, British Report DEG-Memo-663, March 1960.
4. J. T. Mihalczo and J. J. Lynn, in Neutron Physics Division Annual Progress Report for Period Ending September 1, 1960, USAEC Report ORNL-3016, p. 71, Oak Ridge National Laboratory, Dec. 13, 1960.
5. R. C. Lloyd, Summary Listing of Subcritical Measurements of Heterogeneous Water-Uranium Lattices Made at Hanford, USAEC Report HW-65552, Hanford Atomic Products Operation, June 8, 1960.
6. G. H. Bidinger et al., Nuclear Safety Experiments on Plutonium and Enriched Uranium Hydrogen-Moderated Assemblies Containing Boron, USAEC Report RFP-201, Dow Chemical Co., Rocky Flats Plant, July 7, 1960.
7. C. B. Mills, Minimum Critical Dimensions for Water Solutions, *Nuclear Sci. and Eng.*, 9: 377–390 (1961).
8. W. H. Roach, Computational Survey of Idealized Fast Breeder Reactors, *Nuclear Sci. and Eng.*, 8: 621–651 (1960).
9. R. T. Ackroyd et al., A Unified Theory of Interaction for Two Arbitrarily Mixed Multigroup Arrays Set in Well-Defined and Random Reflectors, *J. Nuclear Energy, Part A: Reactor Sci.*, 11: 57–68 (1960).
10. R. A. Scriven and A. F. Thomas, Neutron Interaction in Fissile Assemblies, in *Progress in Nuclear Energy, Series IV: Technology, Engineering, and Safety*, Vol. 3, pp. 253–291, Pergamon Press, New York, 1960.
11. Regulations for the Safe Transport of Radioactive Materials, International Atomic Energy Agency, Vienna, 1961.

12. W. R. Stratton, A Review of Criticality Accidents, in *Progress in Nuclear Energy, Series IV: Technology, Engineering, and Safety*, Vol. 3, pp. 163–205, Pergamon Press, New York, 1960.
13. A. M. Weinberg and H. C. Schweindler, Theory of Oscillating Absorber in a Chain Reactor, *Phys. Rev.*, 74: 851–863 (1948).
14. J. I. Hoover et al., Measurement of Neutron Absorption Cross Section with a Pile Oscillator, *Phys. Rev.*, 74: 864–870 (1948).
15. F. J. Jankowski et al., Calibration of Control Rods, *Nuclear Sci. and Eng.*, 2: 288–302 (1957).
16. O. C. Kolar and F. A. Kloverstrom, Pulsed Neutron Measurement of Control Rod Worths, *Nuclear Sci. and Eng.*, 10: 45–52 (1961).
17. D. V. P. Williams et al., Army Package Power Reactor Critical Experiment, USAEC Report ORNL-2128, Oak Ridge National Laboratory, Aug. 21, 1956.
18. B. E. Simmons and J. S. King, A Pulsed Neutron Technique for Reactivity Determination, *Nuclear Sci. and Eng.*, 3: 595–608 (1958).
19. B. E. Simmons, The Dynamic Reactivity Interpretations of Pulsed Neutron Measurements, *Nuclear Sci. and Eng.*, 5: 254–256 (1959).
20. R. E. Heineman, *Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1958*, Vol. 12, pp. 650–651, United Nations, New York, 1958.
21. D. J. Donahue et al., Determination of k_∞ from Critical Experiments with the PCTR, *Nuclear Sci. and Eng.*, 4: 297–321 (1958).
22. E. Z. Block and D. E. Wood, Measurements of Negative k_{ex} for a Graphite Uranium Lattice in the PCTR, USAEC Report HW-67179, Hanford Atomic Products Operation, Dec. 1, 1960.
23. W. C. Ballowe et al., Intercomparison Measurements, USAEC GEAP-3639, General Electric Co., Vallecitos Atomic Laboratory, Jan. 18, 1961.
24. P. F. Nichols and R. E. Heineman, PCTR Measurements of Control-Rod Strength, in USAEC Report HW-64866, pp. 105–113, Hanford Atomic Products Operation, Apr. 20, 1960.
25. William J. Price, *Nuclear Radiation Detection*, pp. 258–317, McGraw-Hill Book Co., Inc., New York, 1958.
26. D. Ricabarra et al., Measurement of Fast Neutron Flux in a Reactor by Means of Threshold Detectors, *Reactor Sci.*, 12(4): 177 (1960).
27. A. C. Lapsley, Measurement of High Power Densities Inside Nuclear Piles, *Nucleonics*, 16(2): 106 (1958).
28. P. Mulot et al., Automatic Apparatus for the Measurement of Local Gradients of Neutron Fluxes in a Reactor, *Nuclear Instr. & Methods*, 8: 185–194 (August 1960).
29. W. Schilling, Neutron Flux Measurements at the FRM with a Thermocouple, *Kerntechnik*, 2: 398–400 (December 1960).
30. *Atomic Energy Clearing House*, 6(47): (Nov. 19, 1960).
31. R. G. Affel et al., Level Transducers for Liquid Metals, USAEC Report ORNL-2792, Oak Ridge National Laboratory, Apr. 28, 1960.
32. A Pneumatic Temperature Probe for Gas-Cooled Reactors, *Nuclear Safety*, 2(1): 30–32 (September 1960).

Section IV

PLANT SAFETY FEATURES

Deposition of Radioactive Materials from Gases

The designer of a nuclear reactor must be able to predict reliably the location and form of the entire inventory of radioactive fission products associated with the reactor system if he is to fulfill his obligation to control the hazards produced by these materials. One of the most important processes affecting the location of fission products is transport by gases.

Radioactive fission products may be introduced into a gas system either by accident or by design. For example, if a major reactor accident occurs and the fission products are dispersed in the reactor containment vessel, many of the fission products will remain suspended as gases or aerosols in the air inside the vessel. On the other hand, fission gases are circulated in the normal operation of a gas-cooled reactor that uses unclad fuel elements. Some of the fission products are released continuously into the coolant-gas stream and are carried throughout the coolant system. In both these circumstances the fate of the fission products is of paramount importance. In each case there are locations where deposition is beneficial and other locations where deposition is harmful.

Accumulations of fission products by deposition must be avoided in components that may require maintenance or replacement or in parts of the reactor system to which access is necessary to control the consequences of a reactor accident. However, deposition of fission products from gases in some places can substantially reduce the amounts of fission products delivered to other more sensitive locations and can also reduce the total amount of fission-product activity which may be carried to the outside atmosphere in the event of an accident. Prediction and control of the deposition of radioactive fission products from gases require a knowledge of the factors that influence deposition.

Theoretical Considerations

Although there are many factors that may affect the deposition of radioactive materials, they can be divided into three general categories: (1) the concentration of the material and conditions in its history which affect its physical and chemical form, (2) conditions near the site of deposition that influence the transport of the material through the gas to the surface, and (3) conditions that influence the retention of the material when it arrives at the surface.

Radioactive material in a gas may be in gaseous form or in the form of a solid or liquid aerosol. If in aerosol form, the behavior of the material in the gas is affected grossly by the particle size of the aerosol. Particles of sizes ranging from slightly larger than molecular dimensions up to several tens of microns in diameter are common in gases. Which of these forms will be taken by the fission-product materials depends on the history of the material. Fission products may be dispersed in the gases by evaporation at high temperatures or by mechanical action. Evaporation at high temperatures followed by rapid condensation at low concentrations is conducive to the formation of small-particle-size aerosols that are relatively stable under most conditions. These conditions are common in reactors, and it is to be expected that aerosols will be formed.

Particulate matter composed of ordinary nonradioactive materials can play an important role in a gas system containing fission products.

Gaseous materials, including radioactive fission products, will tend to approach adsorption equilibrium with the surface of the particulate matter suspended in the gas. In some cases a fission-product material that would be expected to be gaseous under one set of conditions can be adsorbed completely on particles if the concentration of the gaseous species is low and the duration of exposure of the particles to the gas is long. In such a case the gaseous fission-product material will behave as if it were in particulate form.

There are several processes by which material in a gas is transported to the surfaces where deposition can occur. Turbulent mixing continuously presents gas and its burden to the surface. The conditions favoring this process are high velocity and large channel diameter. Gravitational settling is enhanced by large particle mass, long residence time, and small vertical channel dimensions. Diffusion processes are enhanced by small particle size, long residence time, and small channel dimensions. Molecular dispersions diffuse faster than particulate dispersions. Thermal diffusion is enhanced where high thermal gradients exist in a gas and by large particle diameter and small channel dimensions. Centrifugal separation is enhanced by large particle mass and high gas velocity with a change in direction of flow. Under some conditions electrostatic fields may influence deposition. An extensive theoretical treatment of the processes affecting the form of materials and their transport through gases is available in the literature.[1]

The retention of radioactive products after they have reached the surface may be by several kinds of forces. Massive particles can be held in place by the force of gravity. Electrical attraction can sometimes hold the particles in place. Van der Waals forces are effective for smaller particles, and, in the extreme case of a molecular dispersion, these forces manifest themselves as physical adsorption. Certain chemical forms of fission products can react chemically with surfaces of many kinds to form a chemical compound that is retained with great efficiency. The various processes by which fission products are deposited on surfaces have been exploited in the design of filters and adsorbers. This subject was reviewed previously.[2] The subject of the present review is deposition in other parts of a reactor system.

Experimental Investigations

The behavior of radioactive materials in gases has interested experimenters since shortly after the discovery of radioactivity. In 1900 Rutherford[3] discovered that there was "emanation" from certain naturally radioactive materials. His investigation of this radioactive gas, now known as radon, and of the radioactive daughters deposited from it, led to the recognition of the decay relations between various radioisotopes. It was observed that the daughters of the emanations were in the form of extremely small particles that remained suspended in the diluent gas.

Most of the recent experimental work on the deposition of radioactive materials from gases has been concerned with radioiodine. This is because of the volatility of iodine and its very low biological tolerance. Several experimenters have observed that iodine vapor exhibits varying properties, depending on the conditions under which it is studied.[4–7] It is very readily adsorbed on a wide variety of surfaces in varying degrees. Chamberlain[6] showed, by experiments with a small wind tunnel, that deposition of a "carrier-free" radioactive iodine vapor was controlled by diffusion across the laminar layer of gas flowing over a surface and was analogous to heat transfer by forced convection. Flat surfaces of copper and of filter paper acted as "perfect sinks" for iodine vapor in trace concentrations. Any surface imperfection that interfered with the laminar flow caused increased deposition of iodine vapor on the surfaces immediately downstream of the imperfection.

Once iodine is deposited, the desorption from the surface becomes important. Chamberlain and his coworkers[4] report that the rate of desorption of trace quantities of iodine at room temperature is generally slow but that it may be more rapid in the presence of stable iodine, which can exchange with the adsorbed radioactive layer. The desorption is not appreciable from copper or charcoal surfaces, even in atmospheres containing I^{127} at a concentration of 3 g/m^3. There was appreciable desorption from aluminum, polythene, and painted surfaces in clean air, and the desorption was greatly increased by the presence of I^{127}.

Probably the most important factor concerning deposition of iodine vapor onto surfaces is the presence or absence of other substances in the gas stream. Experiments by Chamberlain

and Whiffen[7] indicate that, in the presence of condensation nuclei, the behavior of radioiodine depends on the amount of iodine carrier present and the concentration of condensation nuclei. Their experimental system consisted of a large aluminum box (6 m^3) into which iodine vapor and condensation nuclei could be introduced. The condensation nuclei were produced by heating a Nichrome wire white hot, by combustion of gas in a Bunsen burner, or by passing an arc between lead electrodes. Samples of the gas mixture were drawn from the aluminum box through a sampling system consisting of a filter-paper-lined tube, a high-efficiency filter, and an activated-charcoal trap. By observing the distribution of I^{131} activity between the components of the sampling system, some insight into the behavior of iodine vapor was obtained. The experimenters concluded that (1) I^{131} can behave as iodine vapor, even in the presence of large concentrations of nuclei (20,000/cm^3), provided that sufficient carrier iodine is present to give an air concentration of 10 $\mu g/m^3$, (2) in the absence of carrier iodine, the behavior of trace concentrations of I^{131} (~ 0.1 $\mu g/m^3$) deviates from that of iodine vapor and the more nuclei present the greater the deviation, (3) in the presence of lead fumes (10–20 mg/m^3) with particle sizes 0.1 to 1.0 μ, most of the iodine radioactivity is associated with the lead particles, and (4) the effect of condensation nuclei on the diffusivity and adsorption of carrier-free radioiodine may have important implications when methods of removing the activity from gas streams are considered.

Chamberlain and his coworkers[8,9] are also responsible for information concerning the effect of condensation nuclei on the behavior of the decay products of thoron and radon. These decay products (isotopes of polonium, bismuth, and lead) adsorb rapidly on condensation nuclei. The effect of attachment is to reduce markedly the diffusivity of the radioactive nuclide compared with its diffusivity in the atomic or molecular state. Thus the rate at which the radioactivity will diffuse across a boundary layer to a surface is greatly reduced once it is fixed to a condensation nucleus. Retention of radon and thoron decay products in the human respiratory system is greater when they are generated in air free of nuclei and inhaled in the atomic state than it is when they are allowed to become attached to nuclei prior to inhalation.

Brewer[10] has discussed the possible or probable fate of various fission products that may be present in the coolant gas of a gas-cooled reactor. In the event of a slight fission-product leak from a fuel element, the presence of krypton, xenon, cesium, rubidium, barium, strontium, samarium, europium, bromine, iodine, antimony, cadmium, tin, indium, and silver should be considered. Upon cooling of the gas stream as it leaves the reactor, the condensation of a mixture of metals, oxides, and carbides would be expected, along with CsI and CsBr. The surfaces of carbon dust and the cool surfaces of the heat exchangers will probably remove the remaining nuclides, with the exception of krypton, xenon, cesium, rubidium, iodine, bromine, and antimony. These nuclides may act as permanent gases to a considerable extent.

Information on the deposition of mixed fission-product radioactivity is provided by Conn et al.[11] Fission products were produced for study by uranium fission and were allowed to escape into the cooling-air stream by direct recoil. Their deposition was then examined in the ducting of the experimental system. The particular isotopes and corresponding decay chains studied were I^{131}, Ba^{140}, Ru^{103}, Sr^{89}, and Te^{132}. The test system allowed the air to pass over the fission-product-generating source only once. It was then routed through stainless-steel ducts, through a large charcoal trap, and into the reactor off-gas system. The percentage of iodine, barium, and strontium plate-out differed in the three tests, as well as the amounts of fission products released. According to the data presented, the fraction of activity deposited in the ducting depended only on the amount of the activity released from the fuel. No dependence of deposition on air flow was noted. The percentage of barium and strontium deposition increased as the amount released increased, whereas the percentage of iodine deposition decreased. The air supply was not characterized, and no information concerning the presence or absence of condensation nuclei is available.

Conclusions

Deposition of radioactivity from the gas phase as it occurs in gas systems associated with reactors is complex, and probably each such system will behave differently, depending on the concentration and identity of the various isotopes released from the nuclear fuel; the

presence or absence of other matter, such as dust, condensation nuclei, and water vapor; the temperature of the system; and the construction materials used in the gas system. Deposition of radioactivity from the gas phase is becoming increasingly important in nuclear technology, but, as yet, insufficient experimental evidence exists to permit reliable predictions of the behavior under conditions existing in nuclear reactors.

(R. E. Adams and W. E. Browning, Jr.)

Effects of Blasts on Containment Vessels

A containment vessel for a nuclear reactor must be a safeguard against the effects of a maximum credible accident. The maximum credible accident has been defined for each of the 13 nuclear reactors built prior to 1959 and is discussed in a report prepared by Van Kessel.[12] This report provides an excellent accumulation of the literature (through the latter part of 1958) on reactor safety in general and on the containment vessels for nuclear plants which were in existence or in the planning stage at that time. It lists over 500 references divided into 24 categories; a few of the references are listed in more than one category.

A maximum credible accident is frequently some kind of explosive reaction, and hence the designs of some containment shells must be based on the response of the structure to dynamic, as well as static, pressure loading. In view of the very limited knowledge and understanding of the cause (the explosion) and the effect (the response), initial studies and investigations had to be made for each specific plant. A series of general studies of this behavior, not associated with any specific reactor design, has been made only during recent years. Some of these studies were reviewed previously,[13–15] and some recent results are discussed here.

Experimental Blast Effects

In some reactors the maximum credible accident can be caused by a sudden loss of coolant which, in turn, may cause a rapid pressure buildup in the container. The pressure buildup in a vapor container during loss of coolant was studied experimentally by the Mine Safety Appliances Research Corporation.[16] The tests involved venting, through an orifice, water that was at a temperature of 500°F and under a pressure of 2000 psig from a 512-gal pressurizer into a 10,000-gal tank. Seven runs were made using four orifice sizes and two water-to-air volume-expansion ratios. The data from this limited number of tests are presented in the report,[16] and, as the authors of the report recognize, are insufficient to draw any basic conclusions that are sufficiently definite for future design purposes.

Experimental measurements of the air-blast effects caused by the bursting of simulated core vessels with differing rates of energy release have been reported by the Ballistic Research Laboratories (BRL).[17,18] In these tests stainless-steel cylindrical pressure vessels were used to simulate the reactor cores. The test vessels were burst by employing one of three methods of internal energy release: (1) suddenly releasing a gas under high static pressure, (2) igniting a confined propellant, and (3) detonating an explosive. The vessels were supported 5 ft aboveground, and measurements of the resulting air-blast loading were recorded for various distances from the vessel. In the first series of tests, the vessels were filled with air. The experimental set of data for this series of tests is presented in the first report.[17] From this set of data it was concluded that the blast loading was relatively independent of the rate of energy release within the vessel. It was established that the test data from the energy source surrounded by the pressure vessel could be reasonably well predicted by assuming that the source was equivalent to a certain weight of bare explosive charge, i.e., an explosive which was not discharged in a vessel. The second report[18] presents the results of additional tests that were similar except that the pressure vessels were filled with water instead of air. The pressure-time histories from these tests varied from those of the bare-explosive tests to the extent that it was not possible to establish an "equivalent weight" of bare explosive, as was done for the previous set of results.

Somewhat similar comparison tests by the Stanford Research Institute (SRI) are discussed in two reports[19,20] which were reviewed previously.[15] Simple, scaled reactor models were used in which scaled energy sources (simulating possible nuclear excursions) were released. In this study, energy sources employing high ex-

plosives alone and high explosive–pyrotechnic mixtures were developed to simulate reactor excursions with periods in the 1- to 5-msec range. In most cases the tests[19] were run in pairs with the same or similar energy sources but with varying amounts (75 and 100 per cent) of water in the pressure vessel.

The findings and conclusions from the SRI tests appear to contradict the findings and conclusions from the similar tests made at BRL.[18] The similarity in the two sets of tests is in the comparison of the results from the bursting of containers that are first filled with air and then with water. The findings and conclusions from the series of tests at BRL on water-filled vessels are given in the following quotations from the report.[18]

> Even though the pressure-time history does vary from those generated by bare explosives, the pressures and impulses measured were never larger than those of the air-filled vessels. The vessel fragments as seen from the fragmentation firings had a very low velocity and little penetrating power. Therefore, it may be concluded that the data for air-filled core vessels, which are applicable to the equivalent weight concept, yield an upper bound of pressure and impulse measurements for its water-filled counterpart and can be used as a method of simulation of nuclear reactor runaway to establish an upper bound for blast loading on containment structures from failure of water-filled core vessels.

On the other hand, the findings and conclusions from the SRI tests were the following:[19]

> In the models tested with vessels filled full of water, the velocity of the plug rise was considerably greater in every case than in similar tests where the vessels were only ¾ full. However, in models ¾ full of water, damage to the lower portion of the cylinder was very extensive and the upper portions broke into relatively large chunks. Concrete fragmentation in the models 100 percent full of water was extensive but more generalized over the whole concrete face. It is evident that the air void in a partly full vessel tends to prevent energy transfer to the concrete surrounding it.

It is considered by Davenport and Huber[21] and by Minor[22] that the tests were not sufficiently similar to justify a direct comparison of the experimental findings. Davenport and Huber[21] claim that the adding of the water in the BRL vessels (the vessels contained only air in the initial tests) lowered the gas temperature sufficiently to lower the air shock wave after it expanded to about 200 times its initial volume. This effect of lowering the gas temperature was not so predominant in the SRI tests[19] because the vessels were always filled at least 75 per cent with water. They further point out that the time of the explosive event in the two sets of tests was considerably different; in the BRL tests the entire event was completed in approximately 10 msec, whereas in the SRI tests the event was completed in approximately 200 msec. Minor points out[22] that the effects of the BRL tests[18] were measured at a considerable distance from the bursting vessel. Furthermore, it must be recognized that the containers in the BRL tests were of stainless steel (a ductile material), whereas the containers in the SRI tests were of concrete (a brittle material). Also, the technique that was used for measuring the blast effect was different for each set of comparison tests; piezoelectric air-blast gauges were used by BRL, whereas SRI observed the extensiveness of the shattering of the concrete.

On the basis of the experimental data presented in each of these reports, it appears that the design criteria for the containment vessel are a function of the material and the type of construction. Hence caution must be exercised in extrapolating either of the two sets of conclusions to any great degree.

Theoretical Treatments of Blast Phenomena

The development of theoretical methods for the study of pressure in explosivelike waves has a long history. Prior to World War II, the development was slow and limited; the need was not great, and the equations that described such phenomena were too complicated to be dealt with directly. The advent of high-speed computers has opened many possibilities for obtaining approximate solutions of specific problems. The general solution of the complex partial-differential equations that apply to most cases is still unknown; however, considerable material in the literature during recent years provides methods for obtaining the approximate solutions of specific problems. The basic equations and a method for solving problems of pressure waves in ducts have been published by Rudinger.[23] In this text it is assumed that all flow variables are uniformly distributed over the cross section of the duct, and hence time

and one space coordinate are the only independent variables. The solution of a problem of this type is obtained by graphical and numerical approximations. Complete wave diagrams for the solution of a number of problems are presented in the text.[23] Because of the complexity of the basic equations, most of the work deals only with systems that require just one space coordinate. The numerical solutions of spherical blast waves have been published by Brode.[24]

Another major difficulty inherent in the solution of the basic blast-wave equations is the discontinuity at the shock front. To overcome this difficulty, Brode[24] assumed an equation for the viscosity; he gives credit to previous authors for the method employed.

The response of structureal materials to blast waves is another complicated aspect of the entire problem. The basic theories and problems involved in this study are presented by Rice et al.[25] A detailed discussion of the problems in experimentation with strong shock waves and the various forms of the equation of state which have been assumed are also given in the article.[25]

Reports on theoretical and experimental studies of blast waves and the compressibility of materials have been presented by the Armour Research Foundation.[26–28] The first report[26] is on a technique for studying the behavior of porous material (such as wood) subjected to compressive impact loads. The second report[27] is a theoretical study of the growth of a point-source explosion (spherical blast waves) in an infinite solid or liquid medium. An expression relating pressure and specific volume (equation of state) of the material is assumed. This equation contains three constants that must be determined from experimental tests on the material (compressibility data). The basic equations were solved numerically, and a parametric study of the influence of these constants was made. The results are presented in the report.[27] The results are for highly idealized physical situations, but they do provide information on the relative effectiveness of the materials as shock absorbers. The third report[28] describes the experimental shock-tube testing of three materials: Celotex, redwood, and sugar pine. Points on the pressure–specific volume curve are obtained from the experimental data for each material and are given in the report.[28]

A method of analytically determining pressures, radii, and velocities in all regions of an exploding reactor is presented by Levedahl and Howerton.[29] Their solution is obtained by numerical integration of differential equations relating rate of energy input, pressure, volume, acceleration, and velocity of several concentric shells representing the core, reflector, pressure vessel, shield water, and shield tank. Complex reactors are transformed into equivalent spheres for analysis, and then the results are transformed back to the true configuration. In this report[29] it is claimed that the velocities of missiles resulting from the metal-water reaction are only fractionally higher than the velocities from an explosion resulting from fatigue failure at ordinary pressures.

Conclusions

The present studies of blast waves and the response of materials thereto are, for the most part, highly specialized and idealized. The additions to the literature during recent years have contributed much to the knowledge and understanding of these phenomena; however, much more work (both experimental and theoretical) must be done before the present ignorance factor in the design of containment vessels for nuclear reactors is reduced to any great degree. (F. J. Stanek)

Beryllium Hazards in Nuclear Operations

Beryllium and its compounds are the most toxic of the nonradioactive components of nuclear reactor systems. These materials require special handling during operations that precede reactor startup if harmful effects to personnel are to be prevented. After beryllium and other chemical poisons become radioactive by exposure to neutrons in a reactor, the chemical-poison aspects of materials-handling hazards become of secondary importance and can be adequately dealt with by procedures for handling radioactive materials, as pointed out previously by Grimes.[30] Developments in beryllium containment and monitoring methods are discussed here. The toxic effects of beryllium have been thoroughly discussed by previous reviewers[30,31] and are too well known to require further emphasis. A bibliography[32] of articles on beryllium toxicology is available.

Beryllium Monitoring

One of the limits of air pollution recommended by the AEC is worthy of comment. It is stated that "no personnel should be exposed to a concentration greater than 25 μgms./m^3 for any period of time, however short."[33] It seems safe to say that no installation handling beryllium is in a position to guarantee that it is meeting this requirement, which should be modified to eliminate ambiguity. The American Industrial Hygiene Association[34] has adopted a more meaningful upper limit for transient conditions of 25 μg/m^3 for less than 30 min. Monitoring for airborne beryllium is usually performed by drawing a measured volume of air through a filter paper, determining the total amount of beryllium collected, and calculating the concentration per cubic meter of air by assuming a uniform rate of collection over the sampling period. If, for example, 50 μg of beryllium were collected from 100 m^3 of air, the calculated average concentration of 0.5 μg/m^3 would be well below the acceptable upper limit of 2 μg/m^3 for the average concentration during an 8-hr working day. If, however, all the beryllium had actually been collected from 1 m^3, the concentration would have been 50 μg/m^3 and thus would have exceeded the AEC limit of 25 μg/m^3.

Several investigators have reported efforts to develop instruments that would continuously monitor air for the detection of beryllium. These instruments would give a quick indication of the presence of an unsafe concentration of beryllium in air and would, by shortening the sampling time, come nearer meeting the "however short" time requirement than the usual routine monitoring methods. The first instrument of this type, reported[35] in 1952, had several drawbacks, including the requirement of a great deal of skilled maintenance, sensitivity to particle size and chemical form of the beryllium, and lack of means for checking the calibration of the instrument. Other investigators[36] later developed a similar continuous spectrographic air monitor that seems to have overcome most, if not all, of the troubles inherent in the earlier instrument. An improved version of this instrument, reported recently,[37] is designed to be sufficiently portable so that it is easy to bring the sample intake close to potential sources of airborne beryllium. Air is drawn through a triggered intermittent arc, and the intensity of the 3130 A beryllium doublet is monitored, together with the adjacent background, and the ratio of those intensities is recorded on a ratio recorder. Provision is made for frequent calibration by means of an aerosol of known concentration produced by means of an auxiliary spark between beryllium-copper electrodes. The accuracy and sensitivity of the instrument are said not to be materially affected by the chemical form of the beryllium or the particle size. The instrument is fully automatic and rapid in operation. Its limit of detection is 1 μg/m^3 for the normal 20-sec sampling period, and one result is produced each minute. The electrodes last about 8 hr; thus they would only need to be changed once each shift for continuous operation. This type of instrument is now being offered for sale by a British instrument manufacturer.[38] Other commercial continuous beryllium monitors are also available from another British producer[39] and an American manufacturer.[40] The reliability of each of these commercial monitors in continuous operation apparently remains to be established.

The spectrographic determination of the beryllium collected on Millipore filters in routine air monitoring has been discussed by Brash.[41] His method, which employs an aluminum internal standard and a photographic measuring system, is said to be sensitive to 0.05 μg of beryllium per cubic meter of air with a coefficient of variation of less than 11 per cent in the 0.5- to 2.0-μg range. The spectrographic method appears to be widely employed for this type of determination, and it is susceptible to automation[42] when the number of samples justifies the expense. A fluoresence method reported[43] to be in use at Los Alamos Scientific Laboratory is said to be usable in the 0.05- to 300-μg range. Limits for surface contamination have not been established. The fraction of such contamination which may become airborne is more important than that which may become ingested, and this fraction is a function of particle size and density, surface condition, and air velocity past the surface. A method for measuring this fraction under controlled conditions was described in a recent report.[44]

Control of Beryllium Hazards

Containment techniques employed at various installations where beryllium work is in progress have been discussed by several authors.

Equipment and methods used with beryllium in some locations under the most hazardous conditions involving manipulations with dry, finely divided beryllium or beryllium compounds are almost identical to those used to assure the safety of personnel working with plutonium. This is perhaps not surprising because a number of installations have been involved in handling both materials, with the development of plutonium-handling methods preceding, in general, beryllium-handling developments.

Mitchell and Hyatt[43] have described types of enclosures employed at Los Alamos for various machining operations with beryllium, together with their monitoring techniques, air-cleaning system, and housekeeping measures. A review of five years of operating experience is given from the health-physics standpoint.

Lindeken and Meadors have discussed[45] the philosophy governing the control of beryllium hazards at the Livermore site. They state that dry machining and grinding operations, beryllium welding, and processing of BeO in the powder form should be performed in total enclosures. They also say that considerably more care is required for partial enclosures where the area of the opening is adjustable than for total enclosures. Their observations indicate that operators tend to leave too large an area open during hazardous operations, with a resulting low face velocity that endangers both the operator and his coworkers. Respirators are worn only in special cases of work not conforming to usual enclosure standards or in cases of accidental failure of an enclosure. They recommend use of smooth floor coverings, such as linoleum, to facilitate cleaning.

Breslin and Harris discuss[46] in some detail items that they term engineering tools required to control atmospheric contamination. These are (1) the selection of process components that will release the smallest possible quantities of contamination, (2) application of local exhaust ventilation at all points of potential contamination release, (3) careful operating procedures, (4) thorough housekeeping, and (5) provision of suitable personal protective measures. A recent book[47] on beryllium contains a chapter on *Health Hazards and Their Control*. Mention is made of exceptions to the usual control limits as follows:

> If the average in-plant concentration over three months is >2 but < 5 $\mu g/m^3$, or if a single short-term sample is >25 but < 100 $\mu g/m^3$, work can continue if operators wear approved respiratory equipment but corrective measures must begin at once. If the three-monthly in-plant concentration exceeds 5 $\mu g/m^3$ or if the single sample exceeds 100 $\mu g/m^3$, work must cease until corrective action is completed.

A brief but fairly thorough coverage of principles involved in assuring the safety of personnel handling beryllium is given.

Two recent reports describing laboratory facilities for experimentation with beryllium are available. Thys[48] describes a small enclosure containing a vacuum glove box for beryllium welding and a glove hood for chemical operations with beryllium compounds. This enclosure is maintained under a negative pressure, and both inlet and outlet air filters are provided, as well as filters for the exhaust air from the glove box and hoods. A clothes-change area equipped with a shower is provided within the enclosure. This facility provides an excellent example of a method for providing secondary containment of hazardous materials and appears to meet the most stringent requirements for a plutonium-handling facility, except for the fact that the gloves are apparently removed from the vacuum glove box when it is evacuated. Most vacuum glove boxes are designed to permit evacuation with the gloves in place, which is highly desirable when the box contains hazardous material.

Aitken[49] describes measures employed to convert an existing laboratory room into a beryllium laboratory. The exhaust capacity of the existing hood was increased by installing a larger fan motor, and both roughing and absolute air filters are provided for the exhaust air. A glove box is attached to the hood through an entrance port so that operations involving the transfer of powdered BeO, weighing, and sintering can be performed in the glove box while other operations of less hazardous nature are carried out in the hood, which has a face velocity of 150 ft/min with the hood opening at the working level. Flexible exhaust hoses are provided in the vicinity of sintering furnaces to pick up fumes or dust, and a Lucite canopy equipped with gloves is employed to keep splatter or dust from a metallographic polishing wheel from becoming airborne.

Conclusions

There appears to be a reasonable probability that one or more of the commercially available

instruments for continuous monitoring of airborne beryllium will solve the problem of determining the extent of air pollution over short time periods. It seems unlikely that these expensive instruments will replace the present air-monitoring methods, which are undoubtedly adequate for most situations. Extensive experience in a number of installations indicates that reasonable air-pollution limits can be met by application of well-established control methods. Although total enclosure of certain types of hazardous operations with beryllium is certainly desirable, and perhaps essential, partial enclosures used with adequate precautions have served satisfactorily for less dangerous work. (C. J. Barton)

References

1. H. L. Green and W. R. Lane, Particulate Clouds: Dust, Smoke, and Mists, E. and F. N. Spon, Ltd., London, 1957.
2. Removal of Fission Product Activity from Gases, *Nuclear Safety*, 1(3): 40–46 (March 1960) and 1(4): 47–53 (June 1960); Removal of Radioiodine from Reactor Gases, *Nuclear Safety*, 2(3): 35–38 (March 1961).
3. E. Rutherford, A Radioactive Substance Emitted from Thorium Compounds, *Phil. Mag.*, [5] 49: 1–14 (1900); also Radioactivity Produced in Substances by the Action of Thorium Compounds, *Phil. Mag.*, [5] 49: 161–192 (1900).
4. A. C. Chamberlain et al., Removal of Radioactive Iodine Vapor from Air, British Report AERE-R-3412, July 1960.
5. A. R. Bancroft et al., Tests on Collectors To Remove Iodine from Ventilation Air, Canadian Report AECL-1131 (CRCE-981), Oct. 31, 1960.
6. A. C. Chamberlain, Experiments on the Deposition of Iodine-131 Vapour onto Surfaces from an Airstream, *Phil. Mag.*, [7] 44: 1145–1153 (1953).
7. A. C. Chamberlain and R. D. Whiffen, Some Observations on the Behavior of Radioiodine Vapour in the Atmosphere, *Geofis. pura e appl.*, 42: 42–48 (1959).
8. E. G. Richardson (Ed.), *Aerodynamic Capture of Particles*, pp. 63–88, Pergamon Press, New York, 1960.
9. A. C. Chamberlain and E. D. Dyson, The Dose to the Trachea and Bronchi from the Decay Products of Radon and Thoron, *Brit. J. Radiol.*, 29: 317–325 (1956).
10. L. Brewer, Fate of Fission Product Gases in the Coolant Stream, USAEC Report GAMD-903 (AECU-4536), General Atomic Div., General Dynamics Corp., Aug. 4, 1959.
11. P. K. Conn et al., Isotope Deposition Hazards in Gas-Cycle Reactors, *Trans. Am. Nuclear Soc.*, 2(1): 210–211 (1959).
12. H. F. Van Kessel, Hazards and Missile Survey of Nuclear Reactor Vapor Containment for SM-2, Report APAE-45, Alco Products, Inc., 1958.
13. Blast Effects on Containment Shells, *Nuclear Safety*, 1(3): 52–53 (March 1960).
14. Structural Integrity of Containment Vessels, *Nuclear Safety*, 1(1): 26–29 (September 1959).
15. D. E. Davenport, Penetration of Reactor Containment Shells, *Nuclear Safety*, 2(2): 31–36 (December 1960).
16. G. E. Kennedy and V. K. Heckel, Pressure Buildup in the Vapor Container During Loss of Coolant Accident, Technical Report 70, Report MSAR-60-6, MSA Research Corp., Jan. 22, 1960.
17. R. J. Larson and W. C. Olson, Measurement of Air Blast Effects from Simulated Nuclear Reactor Core Excursions, USAEC Report WASH-747 (BRL-Memo-1102), Ballistic Research Laboratories, September 1957. Also presented in *Nuclear Sci. and Eng.*, 7: 199–209 (March 1960).
18. W. C. Olson and H. Goldstein, Air Blast Measurements Around Water-Filled Simulated Nuclear Reactor Core Vessels, USAEC Report BRLM-1219, Ballistic Research Laboratories, July 1959.
19. N. R. Zabel et al., Containment of Fragments from Runaway Reactor, USAEC Report SRIA-17, Stanford Research Institute, Oct. 30, 1959.
20. N. R. Zabel et al., Containment of Fragments from Runaway Reactor, USAEC Report SRIA-10, Stanford Research Institute, June 8, 1959.
21. D. E. Davenport and G. B. Huber, Stanford Research Institute, personal communication to F. J. Stanek, Oak Ridge National Laboratory, July 12, 1961.
22. E. E. Minor, Ballistic Research Laboratories, personal communication to F. J. Stanek, Oak Ridge National Laboratory, July 13, 1961.
23. George Rudinger, *Wave Diagrams for Nonsteady Flow in Ducts*, D. Van Nostrand Co., Inc., Princeton, N. J., 1955.
24. H. L. Brode, Numerical Solutions of Spherical Blast Waves, *J. Appl. Phys.*, 26(6): 766–775 (June 1955).
25. M. H. Rice et al., Compression of Solids by Strong Shock Waves, *Solid State Phys.*, 6: 1–60 (1958).
26. H. Napadensky and R. Stresau, A Technique for the Observation and Measurement of the Behavior of Porous Materials When Rapidly Compressed, USAEC Report ARF-D132D11-1, Armour Research Foundation, Illinois Institute of Technology, August 1958.
27. T. A. Zaker, Point Source Explosion in a Solid, USAEC Report ARF-4132-6, Armour Research Foundation, Illinois Institute of Technology, November 1959.

28. P. Lieberman, Shock Impingement Experiments on Crushable Solids, USAEC Report ARF-4132-10, Armour Research Foundation, Illinois Institute of Technology, June 1959.
29. W. J. Levedahl and R. D. Howerton, A Method of Analysis of Reactor Explosions, USAEC Report AECU-3645, Knolls Atomic Power Laboratory. Note: The authors present essentially the same material under the following title: A Method for Computing Pressure, Fragment Velocities, and Core Reactivity During Explosion of a Reactor Pressure Vessel, USAEC Report KAPL-M-WJL-1, Knolls Atomic Power Laboratory, Aug. 7, 1956.
30. Reactor Materials as Chemical Poisons, *Nuclear Safety*, 1(4): 56–58 (June 1960).
31. Beryllium Health Hazards, *Nuclear Safety*, 2(2): 42–45 (December 1960).
32. E. D. Hutchinson et al., Toxicology of Beryllium, USAEC Report UR-570, University of Rochester, Atomic Energy Project, Apr. 29, 1960.
33. Recommendations for Control of Beryllium Hazards, Oak Ridge Operations USAEC Procedure Manual, Chapter OR-OX12, Aug. 26, 1954.
34. American Industrial Hygiene Association, Beryllium and Its Compounds, Hygienic Guide Series.
35. W. L. Churchill and A. H. C. P. Gillieson, A Direct Spectrographic Method for the Monitoring of Air for Minute Amounts of Beryllium and Beryllium Compounds, *Spectrochem. Acta.*, 5: 238 (1952).
36. R. J. Webb et al., A Monitor for the Quantitative Determination of Beryllium in the Atmosphere, British Report AERE-R-2868, March 1959.
37. R. J. Webb et al., Monitor for the Quantitative Determination of Beryllium in the Atmosphere, *J. Sci. Instr.*, 37: 466–471 (1960).
38. Winston Electronics, Ltd., Shepperton, Middlesex, England.
39. Glass Developments, Ltd., London, England.
40. National Spectrographic Co., Cleveland, Ohio.
41. M. P. Brash, Spectrogarphic Analysis of Air Samples for Beryllium Contamination, *Appl. Spectroscopy*, 14: 43–45 (1960).
42. C. S. Murray and J. H. Rowan, Automatized Spectrograph for Detection of Beryllium in Paper Samples, Fifth Conference on Analytical Chemistry in Nuclear Reactor Technology, Oct. 10–12, 1961, Gatlinburg, Tenn.
43. R. N. Mitchell and E. C. Hyatt, Beryllium: Hazard Evaluation and Control Covering a Five-Year Study, *Am. Ind. Hyg. Assoc. Quart.*, 18: 207–213 (1957).
44. G. W. Royster, Jr., and B. R. Fish, Studies of Surface Contamination. I. Intercomparison of Methods for Measuring "Removable" Contamination, USAEC Report CF-61-3-39, Oak Ridge National Laboratory, Mar. 8, 1961.
45. C. L. Lindeken and O. L. Meadors, The Control of Beryllium Hazards, *Am. Ind. Hyg. Assoc.*, 21: 245–251 (1960).
46. A. J. Breslin and W. B. Harris, Health Protection in Beryllium Facilities (Summary of 10 Years of Experience), *Arch. Ind. Health*, 19: 59 (1959).
47. G. E. Darwin and J. H. Buddery, *Beryllium*, Academic Press, New York, 1960.
48. P. C. Thys, Liquid Metal Fuel Reactor Experiment: Investigation of Beryllium Welding Techniques for Reactor Port Thimble Joints, USAEC Report BAW-1100, Babcock and Wilcox Co., April 1960.
49. E. A. Aitken, Small Laboratory Operations Using Beryllium Oxide, USAEC Report XDC-60-2-140, General Electric Co., Aircraft Nuclear Propulsion Dept., Feb. 4, 1960.

Section V

CONSEQUENCES OF ACTIVITY RELEASE

Chemical and Radiation Induced Genetic Exposures

By PETER ALEXANDER* (with Editor's Introduction)

At the request of the Joint Committee on Atomic Energy,[1] a series of advanced technological seminars is being conducted by Oak Ridge National Laboratory (ORNL). The purpose of this program is to study the future role of the national laboratories.

One of the most recent seminars was presented by Dr. Peter Alexander during the course of his visit to the Laboratory. Dr. Alexander is a member of the staff of the Chester Beatty Research Institute of England and is interested in the problem of mutation-producing chemicals. His lecture, entitled "Chemical Contamination of the Biosphere," dealt mainly with such chemicals.

The subject of mutagenesis is not new to persons engaged in the atomic industry. Much attention has been given and continues to be given to the problem in an attempt to determine the mechanism of the mutagenic response as well as the magnitude of radiation dose, dose rate, and radiation energy required to induce such a response.

During a series of hearings before a special Congressional subcommittee on radiation,[2] Dr. Hermann J. Muller, Professor of Zoology, University of Indiana, was called to speak on the subject of radiation-produced mutations. Dr. Muller, winner of the Nobel Prize in physiology and medicine in 1946 as a result of his work on the production of mutations by radiation, stated that evidence has in recent years been accumulating for the broad conclusion that the great majority, if not all, of the damaging effects on life and health evoked by ionizing radiation are results of permanent changes produced in the genetic material: the chromosomes or their contained genes.

Nitrogen analogs of dichlorodiethylsulfide, known as nitrogen mustards, were made during World War II for possible use as chemical warfare agents. Because many of the physiologic actions of the nitrogen mustards resemble those of ionizing radiations, they are called radiomimetic poisons. Research has shown that mutagenesis is one of the actions shared by radiation and this group of chemicals.

The question immediately raised by such findings is: Are there other chemicals, more common and frequently encountered in our environment, which are capable of eliciting radiomimetic actions? Dr. Alexander feels that the answer is in the affirmative, although it is not known whether the magnitude of the radiomimetic action from chemicals in the environment represents a real health hazard. His case is presented below, as abstracted from a paper originally published in *The New Scientist*.[3]

(C. E. Breckinridge)

Mutation-Producing Chemicals

For more than 30 years, the recommendations for safeguards against radiation hazards

*Dr. Peter Alexander was graduated in Chemistry at Imperial College of Science, London, in 1941. After work on a variety of wartime problems, he joined a textile firm as Research Manager. Studies on cross-linking agents in the textile field brought him into contact with cancer research because these same agents had radiomimetic properties and were used in the treatment of cancer. In 1950, he joined the Institute of Cancer Research and is now in charge of the Radiobiology Department at the Chester Beatty Research Institute.

have taken account of the fact that small doses or continuous exposure to low-intensity radiation can be harmful without giving rise to any injuries that can be detected at the time. The injuries, in fact, may not become detectable during the lifetime of a person receiving the dose but may be transmitted genetically to succeeding generations. In Britain, the Medical Research Council; in the United States, the National Academy of Sciences; and, internationally, the United Nations have set up committees of independent scientists to assess the possible genetic dangers to the general population from exposure to low levels of radiation, whether from fallout, medical radiography, or any other cause.

Radiation, however, is not the only source of genetic damage. It is now known that genetic effects of exactly the same type can be produced by a large number of chemical substances. Attention has been drawn to this fact by the National Academy of Sciences in a recently published report on the biological effects of radiation. The Academy points out that further information is needed about the genetic effects of "abnormally high temperatures, ultraviolet light, various chemicals, etc."

During the last 15 years, several hundred chemical agents have been shown to be mutagenic, and it is certain that there are many more still to be identified. Nowhere has a systematic search been made, and the addition of new substances to the list is haphazard. Usually a new compound is not tested for mutagenicity unless the investigation of its mode of action requires such a test. Moreover, the number of laboratories all over the world specifically engaged in the study of chemical mutagenesis is small; they are far fewer than those working on genetic effects of radiation. Almost no work has been done on the genetic effects of chemicals in mammals, and proof of mutagenicity is obtained from experiments with microorganisms, seeds, and insects. Nothing comparable to the huge mouse "factory" of the AEC at Oak Ridge, in which a hundred thousand animals are used for just one experiment on radiation mutagenesis, is available for the testing of chemicals.

No clear pattern can be discerned from among the chemicals that are known to be mutagenic. The most potent are the highly chemically reactive substances that have the ability to combine, by a process known as alkylation, with important cell constituents such as proteins and nucleic acids. Substances that have the ability of alkylate under physiological conditions are the epoxides, which have been shown to be highly mutagenic. Epoxides may be formed when foods are irradiated with atomic radiation in order to sterilize them, and this, as well as the presence of peroxides, raises the question whether such foods are really safe to eat, although they pass the general food tests for toxic properties. As yet this problem has not arisen because the radiation doses needed to produce complete sterility are so high that they alter the taste of food.

Besides the alkylating agents, many other mutagens are known, but in their case there is no known chemical activity to which their mutagenic action can be ascribed. There are many ways by which genes can be altered, and this probably accounts for the diversity of mutagenic agents. Among the mutagens studied in most detail are formaldehyde and some peroxides, although the evidence that hydrogen peroxide behaves in this way is not strong. Peroxides that may be dangerous are produced when fats are heated, and it is possible that they are present in the frying pan. Among substances that have been shown to produce mutations are some antibiotics, some hormones, and components of certain weed killers and contraceptives. Usually these are tested with one type of organism only, and it is not clear whether all these substances are universal mutagens as are the alkylating agents.

At the present time it is impossible to calculate whether chemical mutagens introduced by man into his environment constitute a genetic danger, i.e., whether they lead to a significant increase over the normal mutation rate inherent in cellular reproduction. Most authorities agree that exposure of large populations to atomic radiation must not be such as to double the normal mutation rate, since this would be reflected in a serious increase in certain diseases in later generations. Thanks largely to the pioneer efforts of the mouse geneticists at Oak Ridge in the United States, the order of the radiation dose that would double the mutation rate is known. No such information is available for chemicals. Not only do we not know all the chemicals in our environment which produce mutations, but we cannot calculate the dose that is received by the germ cells as a result, say, of inhaling formaldehyde or eating peroxide. It

is obvious that safety measures can never be as rigorously defined as with radiation, but the difficulties do not justify the complete disregard with which the subject has so far been treated.

In industry the situation is particularly unsatisfactory, since the maximum tolerated levels of volatile substances that may be present in the air of workrooms and factories is determined largely by the amount that produces immediate discomfort or obvious symptoms within a relatively short time. The so-called "tolerable levels" that are published in America and used as a guide here for some substances that are already known to be mutagens (such as ethylene oxide, ethylenimine) would produce in the fruitfly a detectable incidence of genetic damage. Exposure to levels of radiation of comparable mutagenicity would be considered completely unacceptable.

It may be that all the possible embryonic and genetic hazards can, on investigation, be discounted one by one. Let us hope so, but instead of merely hoping, let us find out. At the present time these long-term problems are rarely if ever considered by toxicologists and others responsible for industrial hygiene. This disproportionate preoccupation with immediate effects is in striking contrast with the approach of their opposite numbers in the field of radiation protection.

Dependence of Mutation Rate on Radiation Intensity

A series of papers by Russell and his coworkers[4–7] presented evidence that the radiation induction of mutations is dependent on radiation intensity (dose rate), and their later data[8] strongly support the earlier results that showed a lower mutation frequency with chronic gamma irradiation than with acute X rays. Their later findings are summarized in the following quotation:[8]

> The possibility, rather remote to begin with, that the results could be explained in terms of radiation quality (X-rays versus gamma rays) rather than radiation intensity now seems to be ruled out by additional data obtained from experiments comparing mutation rates induced by acute gamma irradiation with those induced by chronic gamma irradiation. These additional data were obtained with the genetic criterion (specific locus mutations) and cell stage (spermatogonia) most cogent for the comparison.
>
> The view that the difference in mutation rates is due to an effect of radiation intensity on the mutation process rather than on cell selection was originally supported primarily by the results obtained with oocytes. The additional data obtained with spermatogonia now also provide some evidence on this point: (1) Where a probable effect of cell selection on mutation rate was found, it worked in the direction opposite to that required to account for the above difference in mutation rates. (2) The mutation rates at 86 r and 300 r, for chronic gamma irradiation of spermatogonia, are not higher than a linear interpolation between the control points and points at higher doses. Thus, there is no evidence of a distortion of mutation rates by cell selection.

Russell et al.[8] reexamined the possibility that cell selection might account for the intensity effect in oocytes. Even when the comparison between the results from chronic and acute irradiation was restricted to follicle stages in which oocyte killing was infrequent, they found that an intensity effect on mutation rate was observed. In reexamining the possibility that the difference in mutation rates obtained with chronic and acute irradiation of oocytes could be due to differences in oocyte stage, they found that, even when the comparison was restricted to oocyte stages that were closely similar, an intensity effect on mutation rate was found.

They concluded[8] that (1) the effect of radiation intensity appeared to be greater in oocytes than in spermatogonia and suggested various possible explanations and that (2) with acute X irradiation the mutation rate was higher in oocytes than in spermatogonia.

Irradiation of Spermatogonia

Irradiated and control male mice were mated to females homozygous for seven autosomal recessive visibles. The offspring were then examined for mutations at the seven loci. Details of the experimental procedure have been reported.[6,9] The new data for chronic irradiation are presented in Table V-1. Data on chronic and acute irradiation are compared in Fig. V-1, in which the solid points represent results obtained with acute X rays at 80 to 90 r/min, and the open points show chronic gamma-ray results (triangles and square, 90 r/week; circle, 10 r/week). The squares represent mutation rates in females; all other points are mutation rates in males. The point at zero dose is the sum for all male controls. Except for one of

Table V-1 MUTATIONS AT SPECIFIC LOCI INDUCED IN SPERMATOGONIA OF MICE BY CHRONIC GAMMA (Cs^{137}) IRRADIATION

Dose, r	Dose intensity, r/week	No. of offspring	No. of mutations at seven loci	Mean no. of mutations per locus per gamete ($\times 10^5$)
0	0	166,147	10	0.86
86	10	56,993	6	1.50
300	90	28,170	5	2.54
516	90	26,321	5	2.71
861	90	24,281	12	7.06

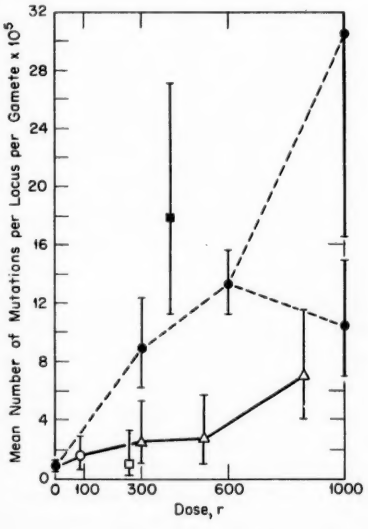

Fig. V-1 Mutation rates of specific loci in the mouse (with 90 per cent confidence levels). Solid points represent acute X rays (80 to 90 r/min). Open points represent chronic gamma rays (Δ and □ = 90 r/week; ○ = 10 r/week). Squares represent females. All other points represent males. (See text for distinction between the two 1000-r points.)

the 1000-r acute dose points, it is clear from Fig. V-1 that higher mutation rates are indicated from acute doses than from chronic doses.

The authors tested their hypothesis that a 1000-r acute dose was killing more of the sensitive cells than was 600 r and that the lower mutation-rate data point at 1000 r was characteristic of the more resistant cells (which were also assumed to have a lower mutation rate than the less resistant cells). The upper point at 1000 r represents a first dose of 600 r, which was followed more than 15 weeks later by a second dose of 400 r. The authors point out the probability that cell selection interfered with the intensity comparison in the case of the single 1000-r dose, instead of accounting for a difference between chronic and acute results, and that the higher divided-dose point is a more valid point for comparison.

It is also pointed out that the chronic doses of 300 and 86 r to spermatogonia do not show evidence of important effects of cell selection on mutation frequencies. Thus Fig. V-1 indicates that cell selection is probably not important in this respect at chronic doses of 861 r or less. A similar conclusion had already been reached in tests of oocytes.[6,7]

Tests of the relative effects of gamma- and X-ray irradiation were continued from previous work[6] and indicated that no great difference in mutation rates was caused by the difference in quality of irradiation. In any case, any effect of quality was much less than the effect of intensity.

Irradiation of Oocytes

The results obtained on irradiation of oocytes are presented in Table V-2 and in Fig. V-1. These data include some obtained in 1959 in experiments[7] in which mutations were measured

Table V-2 MUTATIONS AT SPECIFIC LOCI INDUCED IN MOUSE OOCYTES

Radiation	Dose, r	Dose intensity, r/min	No. of offspring	No. of mutations at seven loci	Mean no. of mutations per locus per gamete ($\times 10^5$)
None*	0	0	5,845	0	0
X*	400	96	1,729	1	17.8
X	400	92	11,124	15	
None	0	0	47,612	0	0
Gamma (Cs^{137})	258	0.0085	27,174	2	1.05

* Experimental work done in 1950 to 1951.

in the offspring of females exposed to acute X irradiation. The mutation rate found was almost 11 times that for chronic irradiation, the latest data for which are also included.

The total plot of the data indicates even more strongly than before that the dose-rate effect on oocytes is on the mutation process itself. Two proposed alternative factors that were rejected as causes of the intensity effect were: (1) cell selection and (2) difference in oocyte stage in the chronic and acute radiation experiments.

Cell selection was discarded as an explanation because, finally, there is a significantly higher mutation rate for acute irradiation even when the chronic irradiation data are restricted to the stages almost certainly free of extensive oocyte killing. (It had been found that the percentage of oocytes killed in certain stages of development was somewhat higher than had been previously noted.[10])

The acute irradiation experiments had been carried out only on dictyate oocytes irradiated in relatively mature follicles, whereas the chronic irradiations involved follicles in a variety of stages of development. The oocytes in early-stage follicles are the more sensitive to radiation killing.[7,11] Therefore an analysis was made in which the only litters considered were those conceived within two weeks following the accumulation of the dose.[7] Combining the results with those of Carter,[12] it was found that the mutation rate for chronic irradiation was still significantly lower than for acute irradiation of females; this indicates that the difference between acute and chronic irradiation is not due to oocyte stage of development. It was noted that this comparison is more rigorous than that in the previous paragraph, where the absence of oocyte killing was the sole criterion for comparability. Additional experiments are being conducted at intermediate intensities, offering better comparisons for closely comparable cell stages.

Comparison Between Results in Spermatogonia and Oocytes

The authors discuss possible reasons for the apparently greater effect of radiation intensity on oocytes than on spermatogonia and suggest that further tests be made to clarify some of the questions raised. The reasons mentioned included (1) somewhat different irradiation "repair" mechanisms, (2) lack of cell selection in oocytes in contrast to the cell selection in spermatogonia which acts to obscure the full effect of radiation intensity, and (3) the existence in acutely irradiated oocytes of a class of mutations not found either in spermatogonia or with chronic irradiation of oocytes.

(W. L. Russell)

River Surveys for Nuclear Power Plants

The wise selection of a site for a nuclear power plant (or a fuel-fabricating or -processing facility) and the judicious operation of the plant or facility after it is built require a study of the possible immediate and long-term effects of radioactive materials on the environment. This is particularly true if any radioactive wastes are to be deliberately released, but the ever-present possibility of accidental loss of control of such materials is reason enough for a careful investigation.

River surveys form an important part of almost all attempts to characterize the environment of a power plant or industrial facility. Whether radioactive materials will reach the river accidentally or through deliberate discharge, the investigator must answer the same primary questions: (1) How rapidly will the radioactive materials be moved to points downstream? (2) In what concentrations will the several radioactive contaminants arrive at these points? (3) How long and in what concentrations will they persist? In addition, prudent planning for waste disposal or for emergency measures in case of an accident requires an evaluation of the possible long-term hazards that might result from assimilation and possible reconcentration of the radioactive materials by the total environment, both physical and biological. This long-term hazard is largely a problem in ecology and is not considered in this review.

Types of River Data Required

Data concerning (1) the volume and velocity of the streamflow, (2) the volume and rate of movement of sediment transported by the stream, both in suspension and as bottom load, (3) the chemical quality of the water, (4) the radioactive-materials content of the water, both natural and from fallout, (5) the temperature of

the water, (6) its microbiotic content, and (7) the degree of turbulence, dispersion, and mixing which result from the flow of the river are all required to answer the primary questions. Because all these quantities or properties are in some degree directly or indirectly dependent on the rate of flow of the stream, all must be studied in terms of rate of streamflow and be related, if possible, to variations in this rate. For almost all streams the studies of variations of stream flow with time and, more particularly, the potential distribution of maximum and minimum flows, present complex problems.

The economic and social importance of rivers has been the subject of many long and penetrating studies, and a variety of valuable methods have been developed for the solution of the complex water problems evolved by personnel of the U. S. Geological Survey and by many others. It is essential that records and data of flow, chemical concentrations, and sediment loads be available which cover a considerable period of time. Reconnaissance surveys must include at least one annual cycle. Comprehensive studies of the distribution and variability of the quantities and qualities of water require, however, a minimum of five years of records and, for many streams, reliable predictions of flow, chemical composition, and sediment concentrations require a base line of many decades. Sustained periods of low rainfall and low streamflow can limit, importantly, the volume of water available to dilute radioactive contaminants at a time when the need for dilution may be imperative. However, a single great flood could move and redistribute river-bottom and riverbank sediment quite beyond the capabilities of any "normal" flood; the sediments thus redistributed would contain concentrated radioactive contaminants.

At Los Alamos, N. Mex., Hanford, Wash., the National Reactor Testing Station (NRTS) in Idaho, and at the small test-reactor site at Dawsonville, Ga., relatively little hydrologic information, at least of the type and reliability required, such as levels of the ground-water table, rates and direction of ground waters, sorption capacity of aquifers, time of water travel in the streams, and deposition characteristics of the streams, was available prior to the initiation of environmental studies. There had been little prior need for these types of hydrologic information at Los Alamos and Dawsonville. The Columbia River, which flows through the Hanford site, had been studied but not in the detail required. In general, rivers such as the Columbia, which drain areas of low rainfall, are subject to larger variations in flow every year, from year to year, and from decade to decade than rivers in more humid areas, and reliable data covering long periods are therefore needed for evaluation of potential hazards.

For the site studies at the Brookhaven National Laboratory on Long Island, some long-term records were available. Water resources of the eastern half of Long Island were studied in some detail in the first decade of this century as a possible source of water for New York City.[13] The records of flow on some of the streams for use in this study were collected in 1903. Continuous records of flow have been collected since 1942 on these streams. However, the gap in the records between the earlier and the more recent work cannot now be filled.

Special Data Required for Radiological Studies

The environmental studies considered here, like studies of flood control, navigation, water supply, and power development, although requiring in each case certain special data, must all be based firmly on a broad understanding of the river system in question. The general and specialized information required for radiological studies has been discussed by Tsivoglou et al.[14] in a paper that is quite useful to those working in the fields of public health and water-pollution control. They describe the organization, planning, and conduct of stream surveys; the basic equipment required for a modest radiological investigation; and the comparatively extensive requirements of the work that would be required to support a comprehensive study. Some details are given concerning the selection of sampling points, frequency of sampling, determination of background radioactivity, and sampling equipment. Equal emphasis is placed on determinations of the levels of radioactivity in the water, the biota, and the bottom sediments.

Types of Hydrologic Stations

Even under rather widely differing circumstances, there will be a need for a network of three distinct types of hydrologic stations: the primary station, the secondary station, and the

partial-record station. Criteria for establishing stream-gauging stations, and the details of their operation, have been described by Grover and Harrington[15] and by Corbett et al.[16]

The purpose of a primary station is to provide a long-period record of streamflow, chemical composition, and sediment loads which is as continuous and complete as is feasible, because only from this long-period record is it possible to statistically describe the variations in quantity and quality of water, both within their normal limits of variation and at times of unusually high or low flow. The period of record required is indefinite, but in many cases a record for a full century would be of material value if it were available. Some long-period, but less than full century, records are available at a limited number of stations on some of our rivers, largely as a result of the work of the U. S. Geological Survey. These data, which are neither as complete nor as accurate as desirable, are published in the annual series of Water Supply Papers.

For many environmental studies, however, there will be no primary-station data available. In general, there will be a station or stations somewhere near the area to be studied, and the records, although falling far short of what might be desired, can provide an invaluable foundation for more detailed local work. It will be a rare and fortunate circumstance when there are detailed long-term data available for a critical point or points within the actual area of interest.

The secondary stations provide a basis for correlations between the primary stations and the principal points of interest in the region being surveyed. In almost every case the secondary stations will be established for the specific environmental survey. Because their value will depend on the length of time their records overlap the records of a primary station, establishment of secondary stations should have high priority. The many problems and uncertainties of the required correlations can be materially reduced if the secondary stations are established at points within areas that resemble, in geology, in physiography, in vegetation, in climate, and in culture, as far as possible, the locations of the primary stations with which they are to be compared. In general, a continuous record is kept at a secondary station only until a correlation with the records at a primary station has been worked out. Usually this will require from 5 to 10 years.

At the third type of station, the partial-record station, only a few measurements are made and only a few samples collected. Where possible, the data are collected at times of maximum or minimum conditions rather than at fixed intervals of time, and an attempt is made to relate these extremes to the conditions at one of the primary or secondary stations.

Types of Correlation Required

The general types of data required, as listed above, include streamflow and stream velocity, the chemical quality and radioactivity of the water, and the nature and quantities of sediment transported. Commonly, there is a relation between streamflow and the chemical quality and sediment loads of the stream. If these relations can be established and the variability and distribution of flow can be defined, the total load transported by the stream can be determined from flow measurements alone. The variations of stream flow should also be studied to determine the percentage of time any given flow rate will be equaled or exceeded, the duration of flow rates, the rate of variation of discharge with time, the magnitude and frequency of floods and of low flows, and the volume of storage required to reduce flood crests or to maintain certain minimum flow rates.

Methods for computing and expressing these relations and for correlating the records of the several types of hydrographic stations have been described by Searcy[17,18] and by Dalrymple.[19] Not only can such correlations be used to predict the volume of water that will normally be available for diluting radioactive contaminants, but also they will make possible estimates of the cost of dams and other structures required to regulate both high and low flows in the stream by storing flood waters and later releasing them. Estimates may also be made of the potential frequency of unusual floods that might damage nuclear power plants or structures of other facilities of the nuclear industry.

Sediment Surveys

A large portion of the radionuclides in radioactive wastes and in the activity that would be released by a reactor accident are cationic and

are strongly adsorbed by the clay and silt particles that form the greater part of the suspended sediment and bed load of most rivers. Indeed, the movement and retention of radioactive contaminants in a river system will largely be governed by the transport of the finer grained sediments to which will be attached most of the fission products. Sorathesn et al.[20] have described some of the factors that affect or control this sorption process. These factors include the chemical character of the radioactive material; the chemical composition of the waste stream and the river; and the composition, grain size, and concentration of the sediment.

The relation of sediment load to stream flow and velocity is complex, and, although the general principles have been more or less defined by a number of classic studies, their application to specific areas is anything but straightforward. Colby and Hembree,[21] for example, have derived a duration curve of suspended-sediment discharge for the Niobrara River in Nebraska from the sediment-flow relation and the flow-duration curve. Similar relations could doubtless be derived for other streams if the basic data were available and perhaps could be extended to define the relation between flow and the distribution of particle size and particle composition of the suspended sediments. Eventually it may be possible to establish similar relations between streamflow and the movement of the bottom load of the stream.[22] Variations in the chemical quality of the river water may also be related to streamflow and to other factors.[23,24] For example, Davis[24] has shown how the quality of the water in a stream is affected by the nature and extent of the geologic formations which underlie the area of its watershed. The natural radioactivity background of the water and sediments in a stream may be similarly controlled by geology; for example, the activity in the sediments in the bed of the Clinch River in Tennessee is highest opposite the mouths of tributaries draining areas of Chattanooga shale, a formation relatively rich in uranium.[25]

Selection of Sampling Sites

Sites for sampling to measure background radioactivity in water and sediments should be established prior to any release of radioactive materials to the stream. Several of these locations should be at the sites established for primary stations on streams to which no radioactive wastes are expected to be released. Background readings at these stations should be carefully correlated with the background in potentially contaminated streams prior to their contamination so that later readings may be adjusted for background on the basis of such correlations. The local geology must be carefully considered in the selection of these sites if the correlations are to remain valid.

Other sampling points should be located at the mouths of large tributaries and at dams, locks, hydroelectric stations, or any other points where streamflows are brought together and channeled so that representative samples may be more easily obtained and the results of the sampling may be correlated with measurements of the streamflow. Where a contaminated liquid waste stream is released to a river, the two most important sampling points are on the waste stream itself, where quantities are easier to measure because concentrations are higher, and on the river upstream from the point of waste discharge, since any activity found there will be background.

For simplicity in operation the first station downstream from the point of release should be located where complete mixing of the contaminant is always assured. In the Columbia River, near Hanford, the contaminants are first detected on the opposite bank 7 miles downstream, and complete mixing is probably assured 40 miles downstream.[26] On smaller streams complete mixing may occur within a few hundred feet of the release point. Another factor that must be considered is the quantity of radioactive material released, for when it is very small, as in the Chalk River,[27] there is little reason to expect to detect the low concentration of radioactivity downstream from the section at which uniform dispersion has occurred. In other cases, however, the contamination may be detected downstream for hundreds of miles, as shown by Nelson.[28] Actually, very little information is available on lateral dispersion in streams, and probably the most effective approach for the determination of the location of complete mixing and for the selection of sampling points is to conduct a series of tracer experiments over a range of flow conditions.

Other sampling points should be located where the first major use is made of the water downstream or where the public has access to the water for swimming or fishing. There are no

general criteria for the selection of such points because they depend entirely on local conditions.

Samples of bottom sediments should be obtained in the vicinity of abrupt changes in flow conditions and intermediate drainage. Abrupt changes in flow may be expected at constrictions and expansions of the river channel, at points where there are changes in channel roughness, at bends, at changes in slope of the river or river-bottom profile, at the mouths of large tributaries, and at dams.

Sample Collection

In general, the collection of some 8 to 15 samples per year at each of the partial-record and areal secondary stations will be sufficient to describe the full range of variation in flows, in chemical compositions, and in sediment-load characteristics, if the times of sampling are carefully chosen to coincide with times of normal flow and with peaks and lows of stream discharge.[29] Care should also be taken to sample the true base flow of the stream. This can hardly be done if the samples are collected routinely, for example, collections which are made on the first day of each month. The visits to the sampling stations have to be irregular and governed by the flow conditions, which may on occasion change quickly with little prior warning.

In streams contaminated by radioactive waste, the practice has been to collect "continuous" water samples. Equipment that is commercially available will collect continuous proportional samples from very small streams.[30] This equipment is placed in a pool upstream from a weir. The creation of such a pool may, however, cause deposition of suspended sediments with their adsorbed radionuclides. If a material balance is to be made, the radioactive materials on these and other bottom sediments must be taken into account.

In large streams the collection of once-daily water samples may be as satisfactory as continuous sampling for the determination of average daily, weekly, monthly, or quarterly concentrations of radionuclides if the size of the daily sample is made proportional to the daily flow and the changes in the flow of the stream and in the rate of release of radioactive wastes within a day are less than 15 per cent. The daily samples may be composited into weekly, monthly, or even quarterly samples, but, if there are many or rapid changes in the streamflow, such composites may not give the true mean concentrations unless the samples are composited in proportion to the flow. Indeed, where variations in streamflow or waste discharge are frequent and rapid, it may well be necessary to take samples for compositing more frequently than once a day. Such samples may be taken, up to about four a day, with automatic programmed timers, sequencing switches, and solenoid valves. If a greater number of samples per day is required from a large stream, electronically actuated continuous proportional samplers should be designed. Such electronic equipment is feasible if the relation between flow and water level for the stream can be determined and if the velocity and flow area can be measured continuously.

Graphs prepared by Colby and Hembree[21] illustrate the degree to which velocity, sediment concentration, and the chemical quality of the water vary from point to point in a stream cross section. Whether samples are collected manually or by continuously operating equipment, they are, in general, taken from the same point. Therefore it is desirable, from time to time, to sample the entire stream cross section adjacent to a sampling point to determine adjustment factors between the mean concentration and the concentration at the sampling point.

Most of the available equipment that has been designed to sample such cross sections is intended to collect only one type of sample, either a sample for determining the quality of the water or a sample for studying the biota or the suspended or bottom sediments; there is no one piece of equipment that can handle all phases of the sampling problem. Dissolved-solids samples may be collected in weighted bottles. Dissolved-solids samples, as well as samples of dissolved gases and very fine suspended sediments, can be collected in Kemmerer samplers (also called Guday and Foerst samplers). Special samplers that tend to minimize disturbance to the flow and so provide more accurate samples have been designed for the collection of suspended sediments, but not all the problems involved in such sampling have been solved.[31] Plankton is sometimes sampled with nets, which tend to clog, and sometimes with Kemmerer samplers.

The greatest difficulty with automatic sampling equipment is to get representative samples of suspended sediments. If the samples

are pumped from the river, a sufficient velocity of flow in the pipeline must be maintained to prevent the settling of sediment; Durand gives data that will aid in the selection of pipe sizes and pump capacities to avoid this source of error.[32]

The bottom load moved by the stream and, indeed, any part of the river bottom, whether stationary or in motion, is particularly hard to sample. Pederson and Ekman dredges will reach only the top few inches of the bottom sediments, and, although other equipment may reach deeper, the investigator must be prepared to design his own equipment to fit his individual needs. If the river scours its bed deeply in times of flood, the radioactive contaminants in the clay, silt, and sand may be distributed through a considerable depth of material. The bottom sediments in many rivers vary greatly from place to place in composition, texture, and cohesiveness. So-called undisturbed'' core samples, although highly desirable if an inventory or material balance of the radioactive materials is intended, are time-consuming to obtain and all too frequently disappointing. There is no one core sampler or standard sampling procedure that is suitable for all bottom sediment conditions. (P. H. Carrigan, Jr., U. S. Geological Survey; approved for publication by the Director, U. S. Geological Survey)

Recent Studies Related to Determination of MPC Values

Since Sr^{90} was first recognized as an environmental contaminant, much attention has been directed toward its metabolic behavior and its distribution in the environment. Data from animal experiments have been extrapolated to obtain data relative to human beings, although it has been shown that, for some nuclides, the biological half life in animals and humans may differ by several orders of magnitude. For example, the problem of calculating the internal dose frequently requires a determination of the fraction of an oral dose which passes from the gastrointestinal tract into the blood. The recent work of Samachson[33] is therefore especially valuable because it provides data from humans that is particularly valuable for calculating MPC values. In his experiments the fraction of the oral dose transferred was 0.20 for Sr^{85} and an average of 0.4 to 0.5 for Ca^{45}. Stable calcium added to the patients' diet was found to have less effect on the amount of Sr^{85} absorbed than on the absorption of Ca^{45}.

Strontium-90 concentrates primarily in the bone, but, in its circulation through the body, it irradiates the reproductive organs and other tissues. Many theoretical calculations have been made to estimate the dose received by the human gonads from both the deposited and circulating Sr^{90}. Samachson found that for a given Sr^{90} intake blood plasma levels of Sr^{90} vary relatively little as between different individuals or from children to adults. However, the concentration of Sr^{90} in the blood plasma does vary directly with the intake of Sr^{90}. For example, on an intake of 5 $\mu\mu$c of Sr^{90} per day, 0.1 $\mu\mu$c/liter is found in the plasma. This results in a dose to the soft tissue, including the gonads, of two or three orders of magnitude less than the dose to the skeleton produced by the strontium absorbed in the bone.

Another fission product that has aroused much interest is Cs^{137}, a gamma emitter, which, unlike Sr^{90}, deposits in the soft tissue rather than in the bone. Under favorable conditions the body burden of Cs^{137} may be determined with a whole-body counter. One of the largest groups to be measured[34] in this way were the women who had worked in the radium-luminizing industry in England prior to the end of World War II. The body burden of Ra^{226} and K^{40} was measured, in addition to that of Cs^{137}. In cases where the body burden of radium was relatively high, the identification of Cs^{137} and K^{40} was appreciably more difficult; so individuals were selected for detailed study who contained less than 0.02 μc RaB in equilibrium with RaC.

The authors conclude from their studies that the Cs^{137} content in different people is not proportional to their body weight nor to their body burden of potassium. In 1958–1959 the mean annual dose received by these adults due to Cs^{137} was less than 2 per cent of the approximate 100 mrem/year received from natural sources.

Another similar study[35] was made of certain residents of the United Kingdom, including those employed at Harwell. The data for Cs^{137} agreed closely with data obtained at Argonne National Laboratory, at Los Alamos Scientific Laboratory, and in Canada. Rundo[35] concluded that the radiocesium found in persons employed at the Atomic Energy Research Establishment at Har-

well was a consequence of world-wide fallout rather than contamination at Harwell and that it amounted to about 1 per cent of the total background radiation. Because of interest in irradiation of the gonads and possible consequent genetic effects, Rundo estimated the dose rate to the testes from Cs^{137} to be about 1.1 mrads/year as compared with 19 mrads/year from K^{40}, an increase of about 6 per cent. Although this represents a possibly significant increase in the background, the total dose is still less than that received by those living in regions of monazite sand. In general, the work of Rundo[35] and of Maycock and his associates[34] provides important confirmation of previous data but offers no new conclusions.

The complexities of environmental contamination have aroused much interest, and many previously apparently unrelated subjects have been combined in the attempt to understand the pathways followed by radioactive fallout. One of the first publications devoted to this problem is *A Symposium on Radioisotopes in the Biosphere*, edited by Caldecott and Snyder.[36] This book is a collection of papers presented at a symposium on the ecology of radionuclides held at the University of Minnesota, Oct. 19–23, 1959. It is the best compilation of data now available dealing with the many facets which must be integrated to understand fallout problems.

Because radioactive environmental contamination is a new and difficult problem with important long-range implications, the U. S. Public Health Service has established a monitoring network in the United States to sample milk, air, and water for radioactivity.[37] The major vehicles by which radioactive nuclides may enter the human body are food, water, and air. After Sr^{90}, Cs^{137}, and I^{131}, which are among the potentially most dangerous nuclides from fallout, are deposited in the body, little or nothing can be done to hasten their excretion, and so quite properly the U. S. Public Health Service wishes to keep track of their presence in food, air, and water. If higher than normal levels are discovered, the source must be sought. If the environmental contamination exceeds the MPC, remedial measures must be taken.

Many of the first experiments in which radionuclides were administered to animals were of necessity of short duration, and only a small number of animals were used. Therefore many of these experiments need to be repeated with a larger number of animals and the work extended to cover a longer period. Because the majority of occupational exposures are from the inhalation of radionuclides, it is particularly important that experiments be done in this area. Interesting work is being done in Russia, judging from the publications of Buldakov and his associates.[38] White male rats were permitted to inhale the chlorides of cerium and ruthenium for 30 and 10 min, respectively, during which periods 40 and 90 μc, respectively, of the nuclides were administered. The animals were sacrificed at intervals varying from 15 min to 64 days. Although these data may be utilized in calculating MPC values, the experiments are still of relatively short duration. Cerium has a biological half life of 1500 days in bone; therefore we may hope that later work will cover even longer periods. (M. J. Cook)

References

1. Joint Committee on Atomic Energy, Congress of the United States, *The Future Role of the Atomic Energy Commission Laboratories*, October 1960.
2. The Nature of Radioactive Fallout and Its Effects on Man, in *Hearings Before the Special Subcommittee on Radiation of the Joint Committee on Atomic Energy, Congress of the United States (85th Congress)*, p. 1065, U. S. Government Printing Office, Washington, June 1957.
3. P. Alexander, Mutation-Producing Chemicals, *New Scientist*, 8: 1073–1074 (1961).
4. W. L. Russell and E. M. Kelly, Comparison Between Mutation Rates Induced by Chronic Gamma and Acute X-Irradiation in Mice, *Science*, 127: 1062 (1958).
5. W. L. Russell and L. B. Russell, *Proceedings of the Second United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1958*, Vol. 22, p. 360, United Nations, New York, 1958. Reprinted in *Progress in Nuclear Energy, Series VI: Biological Sciences*, Vol. 2, p. 178, edited by J. C. Bugher et al., Pergamon Press, New York, 1959.
6. W. L. Russell et al., Radiation Dose Rate and Mutation Frequency, *Science*, 128: 1546–1550 (1958).
7. W. L. Russell et al., Dependents of Mutation Frequency on Radiation Dose Rates in Female Mice, *Proc. Natl. Acad. Sci., U. S.*, 45: 18–23 (1959).
8. W. L. Russell et al., Dependents of Mutation Rate Radiation Intensity, in *Symposium on the Intermediate and Low Level Effects of Ionizing Radiations*, Venice, pp. 311–320 (June 1959).
9. W. L. Russell, X-Ray Induced Mutations in Mice, *Cold Spring Harbor Symposia Quant. Biol.*, 16: 327 (1951); *Genetics*, 41: 658 (1956).

10. E. F. Oakberg and L. B. Blalock, in Biology Division Semiannual Progress Report for Period Ending Feb. 15, 1959, USAEC Report ORNL-2702, pp. 84–87, Oak Ridge National Laboratory, May 14, 1959.
11. W. L. Russell et al., Extreme Sensitivity of an Immature Stage of the Mouse Ovary to Sterilization by Irradiation, *Science*, 129: 1288 (1959).
12. T. C. Carter, Radiation-Induced Gene Mutation in Adult Female and Foetal Male Mice, *Brit. J. Radiol.*, 31: 407–411 (1958).
13. A. C. Veatch et al., Underground Water Resources of Long Island, New York, U. S. Geological Survey Professional Paper No. 44, 1906.
14. E. C. Tsivoglou et al., Stream Surveys for Radioactive Waste Control, in *Second Nuclear Engineering and Science Conference, Mar. 11–14, 1957*, pp. 388–398, Philadelphia.
15. N. C. Grover and A. W. Harrington, *Stream Flow*, John Wiley & Sons, Inc., New York, 1943.
16. D. M. Corbett et al., Stream Gaging Procedure, U. S. Geological Survey Water Supply Paper 888, 1945.
17. J. K. Searcy, Flow Duration Curves, U. S. Geological Survey Water Supply Paper 1542A, 1959.
18. J. K. Searcy, Graphical Correlation of Stream Gaging Records, U. S. Geological Survey Water Supply Paper 1541C, 1960.
19. T. Dalrymple, Flood Frequency Analyses, U. S. Geological Survey Water Supply Paper 1543A, 1960.
20. A. Sorathesn et al., Mineral and Sediment Affinity for Radionuclides, USAEC Report CF-60-6-93, Oak Ridge National Laboratory, July 25, 1960.
21. B. R. Colby and C. H. Hembree, Computation of Total Sediment Discharge, Niobrara River Near Cody, Nebraska, U. S. Geological Survey Water Supply Paper 1357, 1955.
22. B. R. Colby and D. W. Hubbell, Simplified Methods for Computing Total Sediment Discharge with the Modified Einstein Procedure, U. S. Geological Survey Water Supply Paper 1593, 1961.
23. J. D. Hem, Study and Interpretation of the Chemical Characteristics of Natural Water, U. S. Geological Survey Water Supply Paper 1473, 1959.
24. G. H. Davis, Geologic Control of Mineral Composition of Stream Waters of the Eastern Slope of the Southern Coast Ranges, California, U. S. Geological Survey Water Supply Paper 1535B, 1961.
25. W. D. Cottrell, Radioactivity in Silt of the Clinch and Tennessee Rivers, USAEC Report ORNL-2847, Oak Ridge National Laboratory, Nov. 18, 1959.
26. J. W. Healy et al., Flow Patterns in Disposal to the Columbia River, in Sanitary Engineering Aspects of the Atomic Industry, Seminar Sponsored by the Atomic Energy Commission and Public Health Service, USAEC Report TID-7517(Pt. II), p. 23, October 1956.
27. J. E. Guthrie, Radiostrontium and Radiocesium in Ottawa River Water, November 1959–May 1960, Canadian Report AECL-1092 (CRER-962), September 1960.
28. D. J. Nelson, unpublished data.
29. F. H. Rainwater and L. L. Thatcher, Methods of Collection and Analysis of Water Samples, U. S. Geological Survey Water Supply Paper 1454, 1960.
30. H. A. Trebler, Waste Saving by Improvements in Milk Plant Equipment, in *Proceedings of First Industrial Waste Utilization Conference*, Purdue University, November 1944.
31. Subcommittee on Sedimentation, Federal Inter-Agency River Basin Committee, The Design of Improved Types of Suspended Sediment Samplers, Report No. 6, May 1952.
32. R. Durand, Basic Relationships of the Transportation of Solids in Pipes—Experimental Research, Proceedings of Minnesota International Hydrologic Conference, September 1953.
33. J. Samachson, Plasma Levels, Urinary Excretion and Increase in Body Burden of Strontium-90 in Man, *Radiation Research*, 13: 192–204 (1960).
34. G. Maycock et al., Measurements of Cesium 137 in Human Beings During 1958–1959, *Nature*, 188: 355–357 (Oct. 29, 1960).
35. J. Rundo, Radiocesium in Human Beings, *Nature*, 188: 703–706 (Nov. 26, 1960).
36. R. S. Caldecott and L. A. Snyder (Eds.), A Symposium on Radioisotopes in the Biosphere, Minneapolis, University of Minnesota, 1960.
37. Radiological Health Data, Monthly Report, June 1960, PB 161371-3, Vol. 1, No. 3.
38. L. A. Buldakov et al., The Distribution of Cerium and Ruthenium in Rat Organs, *Med. Radiol.*, No. 6 (1960). English Translation: JPRS-5649 (Sept. 30, 1960).

Section VI

CURRENT EVENTS

EBR-I Operating Experience

The Experimental Breeder Reactor (EBR-I) is a liquid-metal (eutectic NaK) cooled, heterogeneous, experimental reactor with a fast-neutron spectrum and a nominal thermal power of 1.4 Mw [0.15 Mw(e)].[1] It was built for, and has been operated under the supervision of, Argonne National Laboratory (ANL), an AEC prime contractor. The EBR-I was constructed at NRTS, Arco, Idaho, and operation began in 1951.

The EBR-I was designed to fulfill two fundamental purposes:[2] (1) to investigate and demonstrate fast-reactor breeding and (2) to demonstrate the feasibility of using NaK as a coolant in power reactors, both on a purely experimental basis. It was designed to be quite flexible,[3] as to both loading and operating conditions, in order to facilitate the accomplishment of its aims. Both of these fundamental purposes had been largely accomplished by 1957 during operation with two core loadings (Mark I and Mark II), as reported by Thalgott,[2] and data and corresponding theoretical models on several aspects of liquid-metal-cooled fast-breeder reactors had been developed. The measured conversion ratio was 1.01 ± 0.05, and no significant difficulty was encountered with the liquid-metal systems. These developments have been described and followed in ANL progress and topical reports through three core loadings (Mark I, Mark II, and Mark III) and are being employed in the design of a fourth core (Mark IV).[4]

The EBR-I consists essentially of an unmoderated reactor coupled to a heat sink. The heat sink consists of (1) a primary NaK coolant circuit that removes heat from the reactor; (2) a secondary NaK circuit that removes heat from the primary NaK; (3) a superheater, boiler, steam turbine, condenser, and economizer (water) circuit that removes heat from the secondary NaK circuit; and (4) an electric generator driven by the turbine. The primary NaK flow is from an elevated constant-level tank into the reactor, down through the inner blanket, up through the core, out of the reactor, through the heat exchanger, and down to a receiving tank. From the receiving tank it is continuously pumped up to the constant-level tank, which contains an overflow to the receiving tank. Further details of the heat-sink system are described by Zinn[5] and Trocki et al.[6]

Description of EBR-I, Mark I, Reactor

The EBR-I, Mark I, reactor[2,3,7] consisted of three principal parts: the core, the inner blanket, and the outer blanket.

The Mark I core was a hexagon approximately $7\frac{1}{2}$ in. high and $7\frac{1}{2}$ in. in diameter (flat to flat). It contained 169 stainless-steel tubes that held four 94 per cent enriched-uranium slugs sandwiched between natural uranium. The stainless-steel tubes extended through the upper shield for handling purposes. There were also slugs of natural uranium both below and above the fuel inside the tube to form part of the inner blanket. An annulus of NaK formed a thermal bond between the fuel and the stainless-steel tube. The total inventory of U^{235} in the Mark I core was 55.675 kg, including 3.875 kg added for operating excess reactivity.

The inner blanket[7] of the EBR-I consists of elements of stainless-steel-jacketed natural-uranium slugs $\frac{15}{16}$ in. in diameter and $20\frac{1}{4}$ in. long jacketed in 0.022-in.-thick stainless-steel tubes. These elements are also provided with extensions projecting through the upper reactor shield for insertion and removal.

An external air-cooled blanket[3,7] surrounds an inner stainless-steel tank containing the core and inner blanket. The external blanket consists

of natural-uranium bricks clad with stainless steel. The blanket is $30\frac{7}{8}$ in. in diameter and 27 in. high and is air-cooled. This cylinder rests on a hydraulic elevator platform and acts as a shim control (by raising and lowering it). An 81-kg block of natural uranium is centered below the inner tank. It is supported by air pressure controlled by valves and instruments so that a power failure would cause the block to drop away from the reactor core.

Surrounding the external blanket[7] is a graphite reflector 19 in. thick, followed by 9 ft of concrete shielding. Six experimental beam holes pierced the concrete shield and graphite reflector.[7] Thermal columns and a "rabbit hole" also provide for experiments.

Twelve 2-in.-OD control rods of stainless-steel-jacketed natural uranium are provided and are designed for vertical motions through holes in the external blanket bricks. Eight are quick-acting safety rods, and four, with controls for more accurate positioning, are for normal operating control. (The relative speeds of these rods figured in the November 1955 partial meltdown of the Mark II core.) The entire outer blanket, and the shield plug on which it rests, was designed to be dropped "quickly" by the hydraulic elevator in emergencies.

Safety Circuits

Of the 49 safety circuits mentioned by Unbehaun,[7] 26 were designed to provide signals indicating abnormal operation, 8 to provide signals indicating danger (and designed to scram the reactor after 2 min without correction), and 15 were designed to scram the reactor immediately. The immediate scrams were designed to act when any one of the following parameters left its specified range: pile period, fuel temperature, coolant flow rate, coolant temperature, elevator hydraulic pressure, radiation levels at designated points, or position of any of four valves. All safety circuits were duplicated to greatly increase their reliabilities. In addition, periodic checks with check lists are made by the operators, and frequent detailed inspections of all equipment and safety devices must be made and reported to the project engineer.

Fuel Inventory and Reactivity Control

Lichtenberger et al. reported[3] at Geneva in 1955 that the operating control rods contained a total reactivity worth of about 1 per cent k, that the safety rods could remove 0.27 per cent k in 100 msec, and the external blanket could remove 8.2 per cent k at a rate depending on an acceleration of a little less than 1 g. They reported that the critical mass was 48.2 kg of U^{235} and that an additional loading of about 4 kg was included to provide for operating control. They also stated that the average temperature coefficient was -3.5×10^{-5} $\Delta k/°C$ from 38 to 200°C and that filling with the coolant added reactivity approximately equal to that added by 2 kg of U^{235} (thus loss of coolant reduces the reactivity). Separate instrumentation was provided for startup, for safety, and for steady operation, and all safety circuits were duplicated, as mentioned above.

Operation of Mark I and Mark II Cores Through 1955

Unbehaun has reported[7] that, when the Mark I reactor was started up in 1951, it was found that radiation levels around the reactor were higher than had been anticipated. An additional 30 in. of concrete shielding was therefore provided, and operation was resumed. In June 1952 a leak of NaK to NaK was discovered in the primary heat exchanger, and the reactor was shut down for repair. During the shutdown, 16 fuel elements were removed for examination and were replaced by new elements. The duration of shutdown was not stated, but it was reported that more than 1500 Mw(t)-hr of operation had been accumulated by Apr. 15, 1953.

The fuel used in the first core was beta-quenched uranium. During operation of the first core, experiments at ANL showed that an alloy fuel composed of uranium and 2 wt.% zirconium was more resistant to irradiation growth and was free of the irradiation-induced surface-roughening characteristic of beta-quenched uranium.[8] Accordingly, in early 1954, a second (Mark II) core was installed with uranium–2 wt.% zirconium alloy in both the fuel and the blanket slugs.

Lichtenberger et al. reported at the 1955 Geneva conference,[3] with respect to Mark I and Mark II operation, that: "The reactor system has proven to be very reliable and has given very good service during $3\frac{1}{2}$ years of operation. . . . This plant is inherently quite stable and is largely self-regulating. . . ."

On Nov. 29, 1955, a partial meltdown of the reactor (second core) occurred during an experiment described by Zinn[9] as a "last experiment" to make measurements and to determine whether fuel-rod distortion would occur with the reactor on a fast, positive period. He explained that the reactor was scheduled to be placed in stand-by in 1956 and that the fuel was to be removed without definite plans for its replacement. So, with the understood risk that some of the fuel might melt, it was decided that the information sought was valuable enough to warrant the risk, and the experiment was carried out. Since a measurement of the temperature coefficient of the fuel alone was desired, the flow of liquid-metal coolant was shut off. Therefore the experiment was carried out on what amounted to a critical assembly rather than an operating reactor. The period was to be reduced to 0.27 sec, and the temperature of the fuel was to be permitted to reach approximately 500°C so that shutdown would have to be initiated at the proper instant and accomplished rapidly. The reactor was equipped with both relatively slow-acting motor-driven control rods and relatively fast-acting shutoff rods. The control operator was to activate the fast shutoff rods on spoken instruction from the observing scientist in charge. Instead, he mistakenly activated the slower-acting control rods. In less than 2 sec the scientist pushed the rapid shutoff button, and, simultaneously, the automatic power-level trips responded to activate the shutoff rods. This delay in reactivity reduction "was sufficient to permit the reactor power to overshoot to a point where the alloying of uranium and steel and melting of the uranium took place." It developed that melting had occurred in 40 to 50 per cent of the core.[10]

The core assembly was removed from the reactor and shipped to ANL at Argonne, Ill., for examination and disassembly. Observations during disassembly and subsequent simulated meltdown experiments indicated that the porous structure formed in the core could have resulted from the vaporization[10] of entrained NaK.

Operation Since 1955 (Mark III Core)

Thalgott reported[2] in 1957 that, in normal operation, the Mark I and Mark II reactors were quite stable and largely self-regulating. He noted, however, that, when the coolant flow through the reactor was changed, a prompt, positive metal temperature coefficient of reactivity was observed. He reported that experiments with the ZPR-III critical assembly indicated a weak Doppler effect in an EBR-I spectrum but that it was too small to account for the observed coefficient. He proposed another possible explanation; i.e., that bowing of the fuel rods toward the core axis occurred because of the increase of the flux and fission rate as a function of nearness to the reactor axis. He proposed that the inward bowing would result in a positive reactivity effect and concluded that appropriate mechanical design could prevent that difficulty. He also reported[2] that a second phenomenon observed in EBR-I was an increasing tendency toward power oscillation when the ratio of power to coolant flow was increased. He pointed out, however, that a sufficiently large, prompt, negative coefficient could override this tendency and so prevent instabilities. He described the features being incorporated in EBR-I, Mark III, which were aimed at prevention of inward bowing of fuel rods and described the design for increased flexibility in coolant flow arrangement and more complete temperature monitoring.

Although the design of Mark I included stiffeners intended to keep the fuel elements a set distance apart (prevent bowing), the problem had not been fully appreciated. Mark II included much stronger stiffeners to prevent bowing,[11] but it was concluded that radial expansion of the end plates, through which the fuel elements extended, had forced some bowing in the elements. Mark III was designed with great care in this respect to make certain that the fuel elements were rigidly prevented from bowing under all conditions of operation. Whereas the fuel pins in Mark I and Mark II were free standing, Mark III pins were close-packed in subassemblies which were in turn close-packed in the reactor core space.

The ANL annual reports[12,13] for 1957 and 1958 confirmed the predictions that the EBR-I with the Mark III core would be stable under normal operating conditions and that "the features of mechanical design responsible for the instability of the Mark II core have been eliminated in the new, more rigid design." Mark III exhibited negative isothermal temperature and power coefficients, and no evidence was found of a positive coefficient associated with an increase in ratio of power to coolant flow rate.

Stability experiments with the Mark III core were continued[14] in 1959. Exchanges of fuel

and blanket rods to increase the temperature gradient across the flow separator gave results indicating that feedback effects originating in the radial blanket are not important. Another test with the stiffening ribs removed from a subassembly of blanket rods showed little or no feedback effects, indicating that bowing of the blanket rods is probably unimportant as a cause of feedback. In another test, removal of stiffeners from fuel rods "indicated rather conclusively that there is an effect completely consistent with the suggestion that the rods bow inward." In all cases studied the results "tended to substantiate the conclusion that a positive component had been added to the power coefficient" (by the removal of stiffeners).

A mathematical model was developed, and it showed "good agreement with reactor behavior." Use of the model to investigate extreme conditions showed instabilities at conditions considereded most unlikely to be present in the reactor. "The principal conclusion is that stability of fast reactors is not a function of nuclear phenomena but is the result of mechanical phenomena which can be controlled by intelligent design."

A final analysis on the stability studies of the Mark III core was summarized[15] in November 1960. No change was indicated in the conclusion that adequate mechanical design prevented a positive coefficient from fuel-rod bowing. It was indicated that a sufficiently large ratio of power to coolant flow would cause instability but that the thermal power would have to be so high (about 1000 Mw) at full flow (10 Mw at one-third flow) that this is only of academic interest. The power coefficient was found to be nonlinear but negative at all powers. The effect of inlet coolant temperature on the power coefficient was found to be real but minor. Low-frequency (e.g., 400 sec/cycle) feedbacks from temperatures of large structures were found. Feedback from the radial breeding blanket, if any, was too small to be detected.

Operation from August 1960 through March 1961 consisted mainly of training reactor operators. The Mark IV core is in various stages of design, development, and fabrication.[4] It is to use fuel elements of plutonium–0.125 wt.% aluminum alloy jacketed in Zircaloy-2 tubing in a further series of experiments.

(L. A. Mann)

Idaho Chemical Processing Plant Criticality Incident of January 25, 1961

On Jan. 25, 1961, at 9:50 a.m., a nuclear excursion of approximately 6×10^{17} fissions occurred in an evaporator within a shielded cell at the Idaho Chemical Processing Plant (ICPP). This was the second criticality incident in the facility; the first incident was reported in a previous issue of *Nuclear Safety*.[16] The January 25 incident is described in the report of an investigating committee.[17] None of the 251 persons present in the ICPP area at the time of the accident received significant radiation exposure. The maximum radiation dose rate at ground level downwind of the facility was 30 mr/hr due to cloud passage. No costs were incurred which were attributable to contamination cleanup, product recovery, or equipment repair as a result of the incident.

The accident occurred while operating personnel were attempting to use air pressure to remove an apparent plug in the evaporator product-withdrawal line. The product evaporator is of the thermosiphon type, with a geometrically safe bottom section and a 24-in.-diameter nongeometrically safe vapor-disengagement section.

Approximately 40 liters of solution containing 200 g of 90 per cent enriched U^{235} per liter had accumulated in the evaporator. Transfer of the evaporator product through a line that extends from the bottom section through a valve and remote head diaphragm pump to a temporary storage tank could not be initiated. It was not known at the time whether this was because of failure of the pump or a plug in the pump discharge. An attempt was made to remove a possible plug in the pump discharge by closing the valve between the pump and the evaporator and forcing approximately 4 liters of water from a 10-gal tank pressurized with 40-psig air through a decontamination line into the suction side of the pump. The 10-gal water tank was located in the plant makeup area. It became apparent that the possible plug had been removed when the liquid-level recorder on the temporary storage vessel indicated an accumulation of approximately 4 liters. The operator in the makeup area was then instructed by voice communication through a pipe chase from the operating area to close the manual valve between the decontamination

tank and the pump suction. A muffled reply from the operator in the makeup area was heard, and thereupon the operator in the operating area turned on the pump and opened the valve between the evaporator and the suction side of the pump. Immediately upon the opening of the valve, radiation alarms sounded throughout the plant. At the sound of the alarms, the plant operations superintendent proceeded to the master radiation panel in the health-physics office, found it almost completely covered with alarm lights, hurried to the end of a corridor to a plant evacuation alarm station, and actuated the plant evacuation alarm; the alarm was actuated within no more than a minute after the radiation alarms had sounded.

The incident apparently occurred because a burst of air from the evaporator product-withdrawal pipeline lifted a portion of the fissile solution from the geometrically safe leg of the evaporator into the nongeometrically safe vapor-disengagement section. Neutron-activation studies of metal samples from the evaporator system have verified that the excursion occurred in the disengagement section. It is supposed that the process operator in the operating area assumed the air pressure in the decontamination and product-withdrawal pipelines had previously been vented through the temporary storage tank after transfer of the 4 liters of water. The investigating committee stated that the postulated mechanism by which the solution was lifted in the evaporator would be more readily acceptable if the manual valve between the 10-gal pressurized tank in the makeup area and the pump suction line had not been closed, inasmuch as a larger volume of air would have been available for expansion. The operator in the makeup area testified, however, that the valve was closed. The available radiation-instrument charts indicate that the nuclear excursion was of short duration (<2 to 3 min) but did not necessarily consist of a single burst. Analysis of the fissile solution indicated that a total of 6×10^{17} (±25 per cent) fissions had occurred.

Evacuation of the building and outside working areas to a parking lot was complete within 5 to 7 min after sounding of the evacuation alarm, except for one person who was working near an air compressor and did not hear the alarm. That person evacuated about 10 min after the alarm had been initiated, and his film badge showed no detectable exposure. Subsequent analyses of about one-half of the employee film badges and pocket dosimeters and of blood sodium samples, together with bioassays and whole-body counting, indicated a maximum gamma exposure of 55 mrem and a negligible internal or external beta or neutron exposure.

Analyses made subsequent to the incident indicated that activity was released to the environment only through the vessel off-gas system, which exhausts to the local plant stack. The vessel off-gas treatment system consisted at the time of the accident of only a small deentrainment tower packed with Raschig rings and a long pipeline to the stack. The vessel off-gas filters were being bypassed at the time of the incident.[18] The first proof of a criticality incident was received approximately 45 min after the incident by the determination of short-lived Cs^{138} on the stack-sampling filter. The radioactive release from the stack did not cause significant personnel exposures and resulted in a maximum radiation reading at ground level downwind of the stack of 30 mr/hr during cloud passage.

The investigating committee concluded that absolute protection against criticality incidents is neither possible nor practical in enriched-uranium fuel-handling plants, since it is necessary to use administrative control in some of the operations and since the effects of an incident in an adequately shielded and contained facility are not disastrous. The committee did, however, make the following recommendations to strengthen the safeguards within the plant:

1. The use of steam or compressed air to clear obstructed process lines is to be permitted only with the specific approval of the Assistant Manager for Operations after the contingencies have been fully evaluated.

2. Communications between safeguard committees and operating personnel should be improved by codification of safeguard committees' actions, providing a full-time safeguard committee chairman, and improving the direction and follow-up in operator training, particularly with respect to infrequent and emergency conditions.

3. Engineering safety reviews should always consider complete systems rather than single items of equipment. In particular, the potential for maloperation or misuse of any part of the system should be considered.

4. Consideration should be given to the redesign of the evaporator system to increase physical protection against criticality.

5. Attention should be given to improving intercommunication between work areas.

6. As a matter of policy, all personnel badge dosimeters should be analyzed immediately following a criticality incident or accidental release of radioactive material.

7. Additional technical personnel should be available each shift to provide increased surveillance over processing activities.

This incident served to emphasize that dispersive incidents are most likely to occur during periods of nonroutine operations, such as the line-unplugging operation, which may be less thoroughly planned and executed than normal process operations. A recommended policy for such nonroutine operations is that no operation should be performed until a technically responsible employee has developed a detailed step-by-step operating procedure, the procedure has been approved by another responsible technical employee, and the procedure has been reviewed and discussed with the personnel who will perform the operation.

A study is being made of the use of a fixed poison to provide additional safety in nongeometrically safe sections of existing process equipment. It is often possible to place glass Raschig rings in evaporator disengagement sections, solvent deentrainment vessels, column end sections, or miscellaneous collection vessels with relatively little inconvenience or cost.

From the relatively minor personal exposures that resulted from this accidental criticality, it is apparent that the containment features of the plant functioned satisfactorily. The exposures could possibly have been even smaller, however, had the evacuation signal been given more promptly, as it could have been if it had been possible to actuate the evacuation alarm from the master radiation panel. Automatic actuation of the evacuation alarm should be considered; this frequently involves the use of coincident signals to avoid unnecessary evacuations.

The reviewer would also like to reiterate the comment that was made in the review of the first ICPP criticality accident,[16] namely, that the consequences of the accident would have been lessened if a more elaborate vessel off-gas system had been used. In particular, off-gases of radiochemical processing plants should pass through a caustic scrubber and then through roughing and absolute filters before being exhausted through a stack. It has been calculated that, utilizing such a system, the maximum possible cloud dose and downwind ground contamination resulting from a criticality incident of 10^{18} fissions would be negligible.

(J. P. Nichols)

Comments on Proposed Reactor Site Criteria

Proposed criteria for use as guides in the selection of reactor sites were published in the *Federal Register* on Feb. 11, 1961, and comments or suggestions addressed to the Division of Licensing and Regulation of the AEC were invited from interested parties. The criteria and previous siting developments were reviewed in the June 1961 issue of *Nuclear Safety*.[19,20] Even if the proposed criteria were adopted in anything like the present form, the judgment of the designer, the AEC regulatory staff, and the ACRS remains vital to the selection of a site, since, as acknowledged by AEC, site criteria cannot yet be stated precisely.

Comments were received from 30 sources, including ACRS, the Atomic Industrial Forum, and foreign, as well as American, groups. The Committee on Reactor Safety of the Atomic Industrial Forum on Mar. 17, 1961, discussed the criteria in a special Forum seminar participated in by 17 of its members.[21] In addition to comments, the Forum submitted,[22] as did Atomics International,[23] a proposed complete revision of the criteria. The selected quotations appearing in the discussion that follows were taken from many of the comments in an attempt to indicate both their range and depth. Although it is intended that the original meaning of the comments be maintained, all are, of course, presented here out of context, and persons interested in greater detail should refer to the original comments. It is anticipated that the Division of Licensing and Regulation will embody many of the comments and suggestions into a revision of the criteria which will then be issued in the *Federal Register*.

General Comments

For the most part the groups who commented welcomed the criteria, since they saw the need for guide lines for the industry. There was, however, much fear expressed that the criteria were too restrictive and that, once they were set forth in the *Federal Register* as regulations and were fixed in the minds of the public and authorities, it would be difficult to make site

decisions counter to the criteria even though such decisions might be warranted by the particular designs. The use of the calculated example (in Appendix A of the proposed criteria) for a serious accident of a hypothetical reactor was a cause of concern. It was felt that too much emphasis was placed on distances to populated centers, to the detriment of the engineering safeguards that can be developed. A sampling of the general comments follows:

> ... use of the term "guides" may lead to some ambiguity as to their intent and purpose ... the proposed guides had been identified by AEC as ... (part) of the *Code of Federal Regulations* . . . .
>
> There was an obvious difference of opinion among seminar participants as to the intended purpose of Appendix A. ...
>
> ... the guides may, through inadverted phraseology, have a possible adverse impact on public opinion even though their intent might be clearly understood by reactor licensees and others in the atomic industry.[21]

> We believe that the guides outlined in the proposed criteria are most desirable for preliminary evaluation of potential reactor sites. . . .
>
> An effort should be made to establish the tradeoffs which can acceptably be effected between engineering safeguards incorporated in the reactor system and the various distances specified in the present guides.[24]

> To my knowledge, there is no definite requirement by the Commission which would prevent an industrial park or suburban housing development from springing up around a reactor, once it was licensed, and effectively abolishing the "low population zone" so carefully calculated as a requirement for issuing the license.[25]

> It is apparent throughout the proposed guide that it was written for a specific class of reactors and a specific type of containment. This is unfortunate.[26]

> Objective (b) of the proposed guides is not clear because the phrases "not normally considered credible" and "the number of people killed should not be catastrophic" are subject to a considerable range of subjective interpretation. In addition, the latter could provoke public alarm without need. . . .
>
> It is recommended, therefore, that the basic objective in the establishment and use of the proposed guides be stated as:
>
> > "Serious injury to individuals offsite should be avoided and the exposure of large numbers of people in terms of total integrated population dose should be low, if any unlikely, but still credible, accident should occur."[23]

> ... the document recognizes tacitly that the ACRS and the AEC as a whole are not as yet completely willing to trade distance for engineering design. This is a reasonable position at this stage in the development of the atomic energy industry and in view of the projected future population growth.
>
> The intent and desire of the ACRS is that this be a very flexible document and that deviations from this document should be allowed in cases where an applicant can show that his design or his siting of the reactor leads to a safe situation. In fact, the ACRS would like to encourage applicants to come in with deviations in those cases in which they believe that a valid reason for deviation exists.[27]

Specific Comments on Guide

Concern was expressed about some special cases in which application of the proposed concept of a "population center distance" would practically exclude power reactors from areas where most authorities feel they will be utilized. Japan feels that its regulations are likely to evolve from the U. S. regulations, and, in Japan, it would almost be impossible to find power-reactor sites based on the proposed regulations because of the high population density. It was also pointed out by several sources that the regulations created an awkward situation for mobile reactor power plants, such as that of the N.S. *Savannah*. The situation with respect to process-steam reactors would also be clouded. Some of the comments on these special cases follow:

> The proposed guides on reactor site criteria published by the United States Atomic Energy Commission on February 11 this year [have] brought about a great deal of repercussion among the related quarters of the Japanese industry. We consider that it is now a sort of world wide common sense that in the field of nuclear energy even domestic legislations cannot be made without influencing or being influenced by the situation abroad.[28]

> Should these figures of exclusion area, low population zone, and population center distance as appearing in the Appendix A be adopted for granted, it would eventually make it practically impossible to find a site in Japan for installing any U. S. type reactor.[29]

> I would like to note that the Commission's efforts to develop industrial process steam reactors also appear to raise questions about the consistency of AEC's proposed guides on site criteria. As you know, it is not economically practical to transmit

steam for long distances. In most cases, it is necessary that the industrial plants using process steam be located closely around the steam-producing plant. The Commission's distance-from-population criteria for reactors seem to us incompatible with the Commission's expenditures of time and money for the development of process steam reactors. . . .

There seems to be another inconsistency, as far as remoteness from population is concerned, in the Commission's attitude in respect to nuclear-powered ships. . . . I recognize that the [N.S.] *Savannah* has been designed and built with special features to enhance the inherent safety of the power plant. . .[25]

Comments on Muliple Reactor Sites

The question of how to apply the criteria to multiple reactors was introduced. Some of the comments were:

Is the power level to be used for determining distances for population the total power level of all nuclear units on site or only the power level of the largest unit?[30]

Where more than one reactor is located at a particular site, the thermal rating used in hazards calculations pertaining to the total installation should lie somewhere between the thermal rating of the largest single reactor and the sum of all reactors present. If there is negligible possibility that an accident in one reactor could cause an accident in another at the same site, then the thermal rating of the largest single reactor should be the value used in hazards calculations.[24]

Comments on Evaluation Factors

There were several suggestions that the emphasis on distance to population centers as a factor used in evaluating reactor sites should be replaced by an emphasis on the safety of the engineering design. Geological and hydrological factors were discussed in relation to the precautions to be taken to prevent radioactive effluents from running into rivers or reaching underground water tables. Criticism and questions pertaining to the maximum emergency radiation-exposure dose for the population were included among the remarks. Some of the comments were:

We view with concern the site evaluation factors. . . . This approach appears to minimize the possible contributions to safety by advances in design and to maximize the value of isolation. We believe this approach unrealistically undervalues scientific and engineering skill, both present and future.[31]

One fundamental point is that the contribution which site considerations can make to the safety of the public is relatively small, when compared to the contribution made by engineering barriers to the release of fission products. . . .

We regard the proposed population center distance factor as technically unjustified and recommend the development of a rating system which factors in meteorology and other environmental factors, in addition to distance. Such an approach provides a much greater degree of assurance that population center distance will make a significant contribution to public safety.[29]

It seems very unlikely that future inland sites can be found which have water supplies which are not used at some downstream point. . . . We would suggest "The plant design must provide reasonable assurance that radioactive liquid effluents cannot accidentally contaminate usable water supplies and prevent use of the water for long time periods. The provisions to prevent accidental releases as well as the importance of the potentially contaminated water must be considered." . . .

In our opinion, some method of evaluating the risk of exposing large numbers of persons should be developed taking into account the integrated man-rem exposure potential based on the proposed plant including its design features, the meteorology and the size, distance, and direction of areas of substantial population.[32]

We do not question the 25-rem whole-body exposure, but we do believe that the 300-rem thyroid exposure is too high. The highest value iodine exposure equivalent that we could justify at the present time would be 150 rem. . . . We suspect that the higher thyroid values were included in the proposed criteria on the basis that exposed persons may be treated in such a fashion as to reduce their thyroid exposure by a factor of 2. However, the use of the higher value would be justified for a particular reactor only if that installation were prepared to, and were capable of, rounding up all the persons which had been evacuated from the low-population zone and getting them to submit to the required treatment. This does not appear to be either desirable or practical.[26]

The most important single new development which is embodied in this document is the fact that it contained a set of dosage limits which can be used in the event of a very unlikely reactor accident. . . .

The site criteria, as published, do not completely define or take care of the problem of either genetic or somatic damage. The problem is recognized and

stated in the document, but answers to this problem do not as yet exist. . . . The Committee [ACRS] believes that the so-called "man-rem" dose has much to offer as a safeguard for the general protection of the population. We went so far, in December of 1960, as to suggest to the Commission that a dose limit of 4×10^6 man-rems per accident might be reasonable.[27]

Comments on Appendix A

The example in Appendix A, which gave calculations for possible reactor siting distances, was the most controversial portion of the proposed regulation. Some of the comments received with respect to the example follow:

> . . . the AEC's Division of Licensing and Regulation has clearly demonstrated a tendency on the part of AEC to take the most conservative of alternative interpretations permitted by any rule. Hence, unless there is an unequivocal intent on the part of the AEC not to permit variance from the assumptions used in Appendix A, it should not be included as an integral part of the guides.[21]

> . . . we also believe that the example at the end should be treated simply as an example and, perhaps, should be either deleted or augmented by the addition of several other examples.[27]

> The proposed criterion tabulates conversion factors which were used to determine the dose received from breathing a specified quantity of the various iodine isotopes. We believe that the actual conversion factors are appreciably higher and suspect that the lower values in the guide were calculated from an earlier inaccurate value of the uptake of iodine by the thyroid as well as a lower breathing range.[35]

> We do not agree with the general applicability or the technical validity of the proposed Appendix A and urge that the Appendix be deleted.[29]

(J. R. Buchanan)

Changes in Licensing Regulations

Parts of the AEC rules and regulations, found in Title 10, *Code of Federal Regulations*, Chapter 1, are frequently amended. Those now being considered for change are discussed here.

Part 2: Rules of Practice

A proposed amendment to the rules of practice would require that, in requesting reactor licenses and waste-disposal licenses, the applicants serve copies of their applications on the chief executive officer of the town or country in which their facility will be located. The requirement would apply to power demonstration reactors constructed under contract with AEC, as well as to licensed reactors.[33]

Part 30: Licensing of By-Product Material

Requirements[34] for exemption of tritium from licensing for use in automobile lock illumination were set forth in the *Federal Register* of Apr. 26, 1961. Both favorable and unfavorable comments have been received since then. Persons in favor usually felt that the exemption should also be extended to other users of similar sources.[35–38] Comments against included statements (1) that an unwarranted precedent would be set and (2) that novelty uses of radioactive materials are being developed when convenient nonradioactive alternates are available.

Part 50: Licensing of Production and Utilization Facilities

Change Without Specific Authorization. Proposed amendments to the change procedure rules were published[40] in the *Federal Register* of Apr. 8, 1961. Comments indicated that the general intent of the changes was good, but it was felt, in general, that there should be additional liberalizations in the rules. The requirement of prompt filing with AEC of a report of every change, test, or experiment was felt to be an unnecessary burden. It was suggested that the AEC technical staff should be able to have ACRS review its judgment without being required to submit the matter to a public hearing. It was felt that operators (with licenses issued prior to the effective date of the amendment) should be allowed to submit proposed specifications if their licenses did not include such, rather than having the entire hazards summary report taken to be the specifications.[35,40–43]

Special Nuclear Material Processing Facilities. An amendment published in the *Federal Register* of June 6 would eliminate the requirement of obtaining a production-facility license if the material processed by a facility contained not more than 10^{-6} g of plutonium per gram of U^{235}. This change will relieve the Englehard Industries, Inc., from the requirement of obtaining a production license in order to process

some fuel and control elements from a Netherlands research reactor.[33]

Part 100: Site Criteria Guide

The comments on this proposed rule are reviewed in this issue under the heading Comments on Proposed Reactor Criteria.

Part 140: Financial Protection Requirements and Indemnity Agreements

On March 29, AEC invited comments on the question of whether processors of unirradiated uranium enriched in U^{235} (not U^{233} or Pu^{239}) should be indemnified and required to furnish financial protection.[40] Since then, many comments have been received. Comments from the nuclear industry have been heartily in favor of the extension, whereas the insurance industry has been wholly against it. The insurance companies feel that this is government encroachment into an area in which they are willing and able to serve.[34–37,44] (J. R. Buchanan)

Action on Reactor Projects by Licensing and Regulating Bodies

The AEC is required by the Atomic Energy Act to protect the health and safety of the public from undue hazards relative to the operation of nuclear facilities. Regulations which are followed to ensure that this responsibility is fulfilled while furthering the simultaneous responsibility for developing the use of nuclear energy are published in the *Code of Federal Regulations*. Title 10 of the Code requires AEC to make certain specific findings regarding the safety of the public before issuing either the construction permit or the operating license for a facility. It also requires authorization for changes in facility equipment or operation which contain an element of hazard not previously reviewed and approved. The license-application record of the various facilities is reported in Table VI-1.

Early in June 1961 AEC sent questionnaires to reactor licensees on gaseous effluent monitoring and control methods to obtain appraisals of their respective practices.[45] The objective of this study is to evaluate: "(1) The capability and accuracy of instruments used to measure selected radionuclides in gaseous effluents; (2) the possibility that significant concentrations of other nuclides may escape detection; and (3) the accuracy of the determination of the character and quantities of nuclides actually released." The results of this survey will be reviewed when they become available.

Recent actions on specific reactors are described below.

Dresden Reactor (Docket 50-10)

On June 5 the Dresden reactor, a 630-Mw(t) boiling-water power reactor, again started supplying power to the Commonwealth Edison Company system and was reported to be operating satisfactorily at a power level[46] of 150 Mw(e). The reactor had been shut down on Nov. 15, 1960, because of a control-rod separation.[47,48]

Following the shutdown in November, the AEC issued, on Feb. 28, 1961, notice of a hearing to consider whether it should authorize the licensee to resume operation with modified control-rod-drive mechanisms and whether it concurred in the proposed test program. Subsequently the licensee filed an application for license amendment to authorize substitution of poison blades of new design, following the observation of cracking of boron-steel blades during modification of the control-rod drive. After hearings were held on these points, AEC granted an interim order on March 31 for limited operation at powers up to 100 kw(t) while review of the new poison blades continued. However, this operation was held up after some cracks were discovered in the reactor core grid structure.[49]

Hearings the last week in April received testimony (from Robert H. Bryan, AEC Division of Licensing and Regulation) which concluded that the grid defects did not significantly impair the immediate usefulness of the structure. Although there was a chance that the defects would increase in number and severity, it was felt that the increase would be gradual. Hence it was felt that the plant could be safely operated if there were adequate surveillance of the structure. It was also concluded that the probabilities of the fallout of a poison blade should be made as low as possible by "following determinations" made as early as possible during the course of a startup. Blade movements should be observed by nuclear instrumentation during other operations with the reactor critical.[50]

Table VI-1 CALENDAR OF LEGAL STEPS IN LICENSING U. S. POWER REACTORS

REACTOR INFORMATION					CONSTRUCTION PERMIT					OPERATING LICENSE				CURRENT STATUS
CODE NO.	NAME OWNER AND OPERATOR LOCATION	REACTOR TYPE	AUTHORIZED POWER Mwt	AUTHORIZED POWER Mwe	ACRS APPROVAL	HEARING EXAMINER APPROVAL	CONSTRUCTION PERMIT NO.	CONSTRUCTION PERMIT DATE	SCHEDULED COMPLETION DATE	ACRS APPROVAL	HEARING EXAMINER APPROVAL	AUTHORIZATION TO OPERATE NO.	AUTHORIZATION TO OPERATE DATE	
50-155	**BIG ROCK POINT** Consumers Power Co. Charlevoix County, Michigan	Boiling Water	157	47.8	Mar. 1960	May 1960			1962					
PP-4	**BONUS** USAEC Puerto Rico Water Resources Authority Rincon, Puerto Rico	Boiling Water Nuclear Superheat	50	16.3	Mar. 1960	June 1960	Not Required		Sept. 1962			Not Required		
None	**BORAX-V** USAEC Argonne National Laboratory NRTS, Idaho	Experimental Boiling Water, Nuclear Superheat	35.7	2.7			Not Required					Not Required		Startup planned in 1962
50-144	**CVTR** Carolinas-Virginia Nuclear Power Associates Parr, South Carolina	Pressurized D_2O Moderator and Coolant	60.5	17	Dec. 1959	Apr. 1960	CPPR-7	5-4-60	Dec. 1962					
50-10	**DRESDEN** Commonwealth Edison Co. Grundy County, Illinois	Boiling Water	630	192 gr 180 net			CPPR-2	5-4-56	Construction started 3-57; completed Sept. 1959	May 1959	Sept. 1959 May 1961	DPR-2	9-26-59 11-16-59 6-2-60 10-14-60 6-9-61	Initially critical Oct. 1959
PP-1	**ELK RIVER** USAEC Elk River Rural Cooperative Power Association Elk River, Minnesota	Boiling Water Thorium Converter	58.2	22	Aug. 1959	Dec. 1959	Not Required		Jan. 1962					
None	**EBWR** USAEC Argonne National Laboratory Argonne, Illinois	Experimental Boiling Water	100	4.5			Not Required					Not Required		Operating since Dec. 1956
None	**EBR-II** USAEC Argonne National Laboratory NRTS, Idaho	Experimental Breeder Fast Neutron Spectrum Sodium Cooled	62.5	16.5			Not Required					Not Required		Startup planned in 1961
None	**EGCR** USAEC Tennessee Valley Authority Oak Ridge, Tennessee	Experimental Graphite Moderated Helium Cooled	84.3	22.3	May 1959		Not Required		Dec. 1962			Not Required		
F-16	**FERMI** Power Reactor Development Co. Detroit Edison Co. Monroe County, Michigan	Fast Neutron Spectrum Plutonium Breeder Sodium Cooled	200	60			CPPR-4	8-4-56 12-10-58 5-26-59	Sept. 1962					On 6-12-61 the United States Supreme Court upheld validity of construction permit
PP-3	**HALLAM** USAEC Consumers Public Power District Hallam, Nebraska	Graphite Moderated Sodium Cooled	240	76	July 1959 Feb. 1960	June 1960	Not Required		July 1962					Hearing on zero-power experiments scheduled for Dec. 1961
50-133	**HUMBOLDT BAY** Pacific Gas & Electric Co.	Boiling Water	163	68.5	June 1960	Oct. 1960	CPPR-10	11-2-60						Submitted final hazards re-

	Consumers Public Power District Hallam, Nebraska	Sodium Cooled			Feb. 1960									Dec. 1961
50-133	**HUMBOLDT BAY** Pacific Gas & Electric Co. Buhne Point, California	Boiling Water	163	48.5	June 1960 July 1960	Oct. 1960	CPPR-10	11-9-60	Aug. 1962					Submitted final hazards report summary in Sept. 1961
50-3	**INDIAN POINT (CETR)** Consolidated Edison Co. of New York Buchanan, New York	Pressurized Water Thorium Converter	585	275 gr 255 net		Final Design June 1961	CPPR-1	5-4-56	Jan. 1962	Nov. 1961				Hearing for provisional operating license scheduled for Dec. 1961
	LA CROSSE USAEC Dairyland Power Cooperative Genoa, Wisconsin	Boiling Water	165	50					1965					
None	**N. S. SAVANNAH** USAEC States Marine Lines Constructed at Camden, New Jersey	Pressurized Water	69	22,000 shp			Not Required			Dec. 1960		Not Required		Loading of fuel and testing of rods authorized 11-24-61
50-130	**PATHFINDER** Northern States Power Co. Sioux Falls, South Dakota	Boiling Water Nuclear Superheat	203	66	Feb. 1960	Apr. 1960	CPPR-8	5-12-60	Apr. 1962					Initial criticality tentatively scheduled for June 1962
50-171	**PEACH BOTTOM** Philadelphia Electric Co. York County, Pennsylvania	High Temperature Unclad Fuel Graphite Moderated Gas Cooled	117	40					Mar. to Dec. 1964	Nov. 1961				Hearing on provisional construction permit scheduled for Dec. 1961
PP-2	**PIQUA OMR** USAEC (Reactor) City of Piqua Municipal Power Commission Piqua, Ohio	Organic Cooled and Moderated	45.5	11.4	July 1959	Jan. 1960	Not Required		Oct. 1961	May 1961				Provisional operating authorization hearing early in 196"
None	**PWR** USAEC Duquesne Light Co. Shippingport, Pennsylvania	Pressurized Water	231	68 gr 60 net	Jan. 1954		Not Required			Nov. 1957		Not Required		Operating since Dec. 1957
50-146	**SAXTON** Saxton Nuclear Experimental Corporation Berks County, Pennsylvania	Pressurized Water	20	3.2	Sept. 1959	Jan. 1960	CPPR-6	2-11-60	Jan. 1962	July 1961	Oct. 1961	DPR-4	11-15-61	ACRS approved startup program including operation up to full power. Initial license is provisional.
None	**SRE** USAEC Atomics International & Southern California Edison Santa Susana, California	Graphite Moderated Sodium Cooled	20	6			Not Required					Not Required		Operating since Apr. 1957
50-18	**VBWR** General Electric Co. Alameda County, California	Test Reactor Boiling Water	50	5			CPPR-3	5-14-56				DPR-1	8-31-57	Operating since Sept. 1957
50-183	**VESR** General Electric Co. Alameda County, California	Light Water Moderated, Separate Steam Superheater	12.5		May 1961	July 1961		8-10-61	Apr. 1962 to Feb. 1963					
50-29	**YANKEE (YAEC)** Yankee Atomic Electric Co. Rowe, Massachusetts	Pressurized Water	485	141	Sept. 1957		CPPR-5	11-4-57		Feb. 1960 May 1960 June 1960	April 1960 July 1960 June 1961	DPR-3	7-9-60 7-29-60 6-23-61	On 6-23-61 Yankee's Provisional License was converted to a 40-yr Operating License

On May 27 Hearing Examiner Jensch issued a Fourth Supplemental Intermediate Decision which recommended issuance of a license allowing Dresden to resume operation at power levels up to 630 Mw(t). It was held that operation of the reactor with the modified drives could be resumed without endangering the health and safety of the public, subject to considerations involving inspections, tests, and operating procedures. In addition to periodic inspection of the control-rod-drive mechanisms, the poison blades, and the grid structure, Commonwealth Edison will test its container and mainstream shutoff valves. The containment tests would be made before Oct. 15, 1961, with followup tests to be made at subsequent intervals. Fourteen new fuel assemblies were approved for the reactor subject to the present license restrictions on burnout margin.[51] The new requirements and specification changes were authorized in an amendment to Appendix A of Edison's license due to expire May 4, 1996.

During startup operations, Dresden found that one of the nine control-rod drives equipped with thermocouples indicated a somewhat higher temperature than the other eight. On May 31 the approach to power was delayed so that thermocouples could be installed on the remaining drives. While exercising the one drive during the installation period, the condition apparently cleared. None of the drives were equipped with thermocouples during the previous operations, but there is now one thermocuple on each drive.[46]

Elk River Reactor (Docket PP-1)

Specific information was requested by the Division of Licensing and Regulation regarding cracks in the pressure-vessel cladding of the Elk River 58-Mw(t) boiling-water reactor.[49] Allis-Chalmers Manufacturing Company submitted a written report[52] to confirm oral information given to AEC on June 2. It was judged by Allis-Chalmers that the cracks in the cladding would have no deleterious effects on base-material corrosion, strength of the vessel, or in the radioactivity levels in the primary loop. Allis-Chalmers also felt that the cracks which appeared after the vessel was pressure tested were caused by release of locked-in stresses. Further crack formation under operating conditions is not expected.[53,54]

Allis-Chalmers also filed a report[55] to update an earlier report of May 4 on the 17-4-PH steel in the reactor. A report on control-rod blades was also submitted[56] on May 4.

Enrico Fermi Atomic Power Plant (Docket F-16)

The Supreme Court, on June 12, reversed the judgment of the lower court and upheld the legality of the AEC's order that granted a provisional construction permit for the Enrico Fermi fast-breeder reactor.[49] On June 10, the Power Reactor Development Company (PRDC) requested that the construction permit and its licence application show Dec. 15, 1961, as completion date for the reactor,[57] rather than July 15, 1961. The extension was granted on July 14 by the Licensing and Regulation Division of AEC.[58]

On July 13 PRDC filed portions of a revised application for an operating license for the reactor. Sought is a 25-year license for reactor powers up to and including 200 Mw(t). Included in the application were six volumes of a seven-volume hazards report.[58]

FWCNG Reactor (Docket 50-152)

As indicated previously,[49] work on a proposed 153-Mw(t) D_2O-moderated gas-cooled reactor was terminated in early May 1961 by the AEC. On July 12 FWCNG was advised that actions were being terminated on their pending construction license application per their request[57] of June 26.

Indian Point Reactor (Docket 50-3)

A hearing was held early in May 1961 on the final design of the 585-Mw(t) pressurized-water thorium-converter Indian Point reactor in which Consolidated Edison Company of New York, Inc., sought approval relating only to design and construction.[49] In an intermediate decision issued on June 30, Examiner Bond recommended approval of the design. The "ultimate conclusion" of the Examiner was made subject to a condition that the applicant present satisfactory evidence on the metallurgical characteristics of the control-rod-drive shafts.[59]

On May 15 Consolidated Edison had their counselors, LeBoeuf, Lamb, and Leiby, file an amendment (No. 4) to the Company's amended and substituted license application, presenting as exhibits operating procedures and preoperat-

ing and startup test procedures.[60] On July 18 they submitted a tentative regulation schedule that will lead to an anticipated issuance of an initial operating license. The schedule included a September date for a meeting with ACRS to discuss all matters pertaining to an operating license and completion of plant construction.[61]

Nuclear Merchant Ship N.S. Savannah

The AEC issued in late July 1961 an authorization for the startup and operation of the 69-Mw(t) pressurized-water reactor of the N.S. *Savannah* through initial sea trials.[62] During March and April, public hearings were held concerning fueling and operation of the reactor for test and demonstration purposes.[48,49] The AEC staff filed its Proposed Findings of Fact accompanied by a Memorandum for AEC Regulatory Staff recommending that AEC authorize operation as proposed.[63]

The AEC authorization incorporated the findings and conclusions from the favorable recommendation of Hearing Examiner Jensch[64] but modified it as proposed in part by the Licensing and Regulation staff in a memo of June 29. The conditions of authorization seek to assure that detailed written operating and maintenance procedures for each phase of the reactor testing program will be prepared before commencement of that phase. Startup tests at Camden, N. J., are limited to no more than 7 Mw. After completion of tests at Yorktown, Va., and initial sea trials, the Licensing and Regulation Division and the ACRS will review all reports on test operations and set down their recommendations in a public hearing before reactor tests are continued or the ship is returned to Camden while pressurized.

Piqua Nuclear Power Facility (Docket PP-2)

The Piqua reactor is a 46-Mw(t) heterogeneous organic-cooled and -moderated reactor being built at Piqua, Ohio.[65] On May 18–20, 1961, the ACRS met to consider the final safeguards report, a supplement, and a proposal for operation submitted by Atomics International. Earlier (on April 20) an ACRS subcommittee had met on the reactor site with representatives of Atomics International and the AEC staff. Based on information presented and discussed, ACRS felt that the reactor could be operated by Atomics International without undue hazard to the health and safety of the public. ACRS suggested that, at each partial fuel reloading cycle (approximately four- to six-month intervals of power operation), an inspection of a typical sample portion of the remaining fuel elements be made and that a report be forwarded to the Division of Licensing and Regulation.[66]

Plum Brook (NASA) Reactor (Docket 50-30)

A provisional operating license was issued on March 14 for the 60-Mw(t) light-water-cooled and -moderated Plum Brook reactor of NASA.[49] Authorization was issued on June 12 for initial core loading. This action followed the finding by the Division of Licensing and Regulation that systems or items needed for full-power operation (but not requiring completion for low-power testing) were built and tested and that modifications required for 100-kw operation were complete.[67]

On June 29, NASA advised of two changes in the reactor. One was a redesigned grid-lifting mechanism and the other was a modification to the manual control section of the regulating-rod amplifier.[68]

Radiation Effects Reactor (Docket 50-122)

The Lockheed Aircraft Corporation on June 8 advised the Division of Licensing and Regulation that it agreed to withdrawal of its application for an operating license for the 10-Mw(t) pressurized-water Radiation Effects Reactor (RER) but that it anticipated a resubmission at some future date.[69] ACRS had recommended in December 1960 and March 1961 that the reactor be operated at full power in its present form only for work essential to the national defense.[48,49]

Saxton Reactor (Docket 50-146)

The ACRS issued a favorable report on the Saxton 20-Mw pressurized-water reactor based on documents considered in its meeting of July 6–8. They noted that there was the possibility that complete withdrawal of one single control rod could make the reactor critical but that a system involving manually set limit switches is provided to prevent excessive rod withdrawal and to provide an adequate shutdown margin. The committee concluded that the reactor could be operated through its startup without undue hazard to the health and safety of the public.[70]

The Saxton Nuclear Experimental Corporation, on June 9 in its sixth license-application amendment, listed proposed technical specifcations, proposed a definition of the Saxton facility, and requested that discussion of Saxton's technical and financial qualifications be considered in a future September public hearing rather than the July hearing.[71]

Vallecitos Boiling-Water Reactor (Docket 50-18)

Permission to resume operation of the Vallecitos Boiling-Water Reactor (VBWR) was granted by the AEC on Apr. 13, 1961, under conditions providing for control-rod testing, inspection, and behavior study.[49] The operator, General Electric Company, concurred with the conditions with slight modifications.[72] On May 22 General Electric, in an "overdue followup," reported on the Division of Licensing and Regulation's concern over General Electric's apparent failure to report cracks found in one of the VBWR boron–stainless steel control rods during a routine inspection of March 10. General Electric reported that there had been no intent to withhold information since there had been a voluntary report of the incident to a San Francisco Operations Office inspector on March 15. It was felt by General Electric that the cracks did not constitute a "possible unsafe condition" since structural stainless steel backs up the boron–stainless steel. On May 16 General Electric sought permission to test a new control blade material by fastening it to a standard blade to test its long-time performance in the reactor.[73]

On June 6 General Electric filed with the Division of Licensing and Regulation reports of four proposed changes or experiments. Three had to do with irradiation experiments; the fourth was on a steam-water separator being installed in the reactor pressure vessel to reduce entrainment of water in the effluent steam.[74] On May 4 General Electric reported on a proposal to irradiate a fuel assembly prepared from rejected *Savannah* core II stainless-steel tubing to study the effects of microscopic leaks and irradiation on the tubing.[72] The Division of Licensing and Regulation on June 15 expressed concern over the fuel assembly and requested a reappraisal with additional information.[75] General Electric submitted an amendment (No. 46) to VBWR's application[76] with respect to a loop called E-SADE on June 26. On July 12 General Electric proposed an experiment to study the causes of a leak in one of the VBWR steam-generator tubes.[77]

Vallecitos Experimental Superheat Reactor (Docket 50-183)

The AEC's Chief Hearing Examiner Jensch in an intermediate decision recommended on July 25 that a construction permit be issued for the 12.5-Mw(t) Vallecitos Experimental Superheat Reactor (VESR) proposed by General Electric.[78] The case had been heard on June 29 when testimony was given by four representatives of General Electric and AEC. Booth of the Division of Licensing and Regulation had testified that General Electric had "supplied sufficient information to provide reasonable assurance that the facility can be constructed and operated without undue risk to the health and safety of the public."[79]

ACRS has considered the reactor in two full committee meetings and two subcommittee meetings. In a report issued after the latest meeting (May 18–20), the Committee concurred in the desirability of the safety rods proposed by General Electric in addition to nuclear poison rods. The ACRS listed six items which they would like to consider at a later date.[66]

On June 19 AEC's Division of Licensing and Regulation held an informal public meeting at Pleasanton, Calif., to discuss its regulatory program and its review of VESR in order to give members of the public living in the area a chance to ask questions. Whether or not this practice will be extended to other reactors is not known.[80]

Westinghouse Test Reactor (Docket 50-22)

On April 25 Westinghouse requested an amendment of its 60-Mw(t) Westinghouse Test Reactor license to authorize them to lower the head tank-to-vent effluent discharge ratio by a factor of 2 while increasing the monitor alarm setting by the same factor.[81] The Division of Licensing and Regulation authorized the change[82] on May 17. Westinghouse submitted results on May 18 of a sampling program on the effluent.[83] On July 14, Westinghouse submitted a report requesting authority to modify a high-pressure test thimble.[84]

Yankee Reactor (Docket 50-29)

In a license amendment dated June 23, AEC authorized the Yankee pressurized-water reactor to operate at power levels[85] up to 485 Mw(t). Yankee's provisional license was also converted to a 40-year operating license. Hearing Examiner Jensch authorized the amendment in a decision[86] dated June 20, following a public hearing[87] on June 8. Both ACRS and the Licensing and Regulation staff concluded that the reactor which had been operating at 392 Mw(t) satisfactorily could be operated at the higher power level without undue hazard to the health and safety of the public. The Division of Licensing and Regulation approved several minor modifications to the plant and specifications designed to improve performance and to correspond with changes. (J. R. Buchanan)

Safeguards Reports and Selected Reading

The recently issued safeguards reports and selected literature pertaining to hazards of reactors are listed below for reference. Because of the similarity of many reactors (in particular, research reactors), this list is not intended to be all inclusive.

1. R. J. Smith (Comp.), Reactor Safety: A Literature Search, USAEC Report TID-3525 (Rev. 3), July 1961.

2. J. Coombe et al., Hazards Report for the SM-1 Core II with Special Components, USAEC Report APAE-84, Alco Products, Inc., Mar. 30, 1961.

3. J. G. Gallagher, Hazards Report for the SM-1 Core II Without Special Components, USAEC Report APAE-84(Add. 1), Alco Products, Inc., Apr. 19, 1961.

4. N. G. Wittenbrock and J. Muraoka, Plutonium Recycle Test Reactor Final Safeguards Analysis, Supplement 2. Consequences of a Primary Coolant Leak, USAEC Report HW-61236(Suppl. 2), Hanford Atomic Products Operation, Nov. 15, 1960.

5. R. W. Clack et al. (Comps. and Eds.), Hazards Summary Report for the Kansas State University TRIGA Mark II Reactor. Special Report No. 7. Secs. I–III, Report NP-10074, Kansas State University, January 1961.

6. Union Carbide Nuclear Co., Research Center, Tuxedo, N. Y., Final Hazards Summary Report for the UCNC Research Reactor, Report NP-9674, November 1960.

7. D. Hunter et al., Yankee Critical Experiments Hazards Summary Report, USAEC Report YAEC-31(Suppl. 2), Westinghouse Electric Corp., Atomic Power Dept., July 1, 1958.

8. J. Carothers, Hazards Summary Report for the LRL Critical Facility, USAEC Report UCRL-6220, University of California Lawrence Radiation Laboratory, Mar. 3, 1960.

9. J. L. Murray, General Electric Vallecitos Boiling Water Reactor, Final Hazards Summary Report, Report SG-VAL-2(3rd ed.), General Electric Co., Vallecitos Atomic Laboratory, Nov. 30, 1959.

10. General Atomic Div., General Dynamics Corp., Hazards Summary Report for Diamond Ordnance Radiation Facility, Report NP-10188, 1960.

11. E. Feinauer and R. D. Thomas, Safeguards Report for the Northrop Pulse Radiation Facility, Report NP-9978, Norair Div. of Northrop Corp., Mar. 22, 1961.

12. A. H. Emmons and W. R. Pearce, Preliminary Hazards Report, Exhibit A, Class 104 Facility License Application with Exhibits B, C, and D, Report NP-9951, University of Missouri, March 1961.

13. P. M. Finnegan, Preliminary Hazards Summary, Mock-Up Reactor, NASA Plum Brook Reactor Facility, Report NP-9947, National Aeronautics and Space Administration, Lewis Research Center, January 1961.

14. P. D. O'Brien, Hazards Evaluation of the Sandia Pulsed Reactor Facility (SPRF), USAEC Report SC-4357A(RR), Sandia Corp., February 1961.

15. C. L. Blackshaw and C. H. Skeen, Safeguards Summary for the AE-6 Reactor, USAEC Report NAA-SR-5499, Atomics International [no date].

References

1. Nuclear Reactors Built, Building, or Planned in the United States, as of June 30, 1959, USAEC Report TID-8200.
2. F. W. Thalgott, Fast Reactor Safety; EBR-I, Proceedings of the 1957 Fast Reactor Information Meeting, Chicago, Ill., Nov. 20–21, 1957, pp. 68–76.
3. H. V. Lichtenberger et al., Operating Experience and Experimental Results Obtained from an NaK-Cooled Fast Reactor, *Proceedings of the United*

Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955, Vol. 3, pp. 345–360, United Nations, New York, 1955.
4. Argonne National Laboratory, Apr. 15, 1961. (Unpublished)
5. W. H. Zinn, Basic Problems in Central Station Nuclear Power, *Nucleonics*, 10(9): 8–14 (September 1952).
6. T. Trocki et al., Sodium and Sodium-Potassium Alloy for Reactor Cooling and Steam Generation, *Proceedings of the United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955*, Vol. 9, pp. 241–251, United Nations, New York, 1955.
7. W. E. Unbehaun, History and Status of the EBR, USAEC Report AECD-3712, Chicago Operations Office, Apr. 15, 1953.
8. J. H. Kittel and S. H. Paine, Effects of High Burnup on Natural Uranium, USAEC Report ANL-5539, Argonne National Laboratory, May 1, 1957.
9. W. H. Zinn, Letter, *Nucleonics*, 14(6): 35 (1956).
10. J. H. Kittel et al., The EBR-I Meltdown — Physical and Metallurgical Changes in the Core, USAEC Report ANL-5731, Argonne National Laboratory, November 1957.
11. F. L. Yaggee, The Manufacture of Centrifugally Cast Reactor Fuel Slugs, Mark-II Fuel Loading for EBR-I, USAEC Report ANL-5724, Argonne National Laboratory, July 1960.
12. Argonne National Laboratory, Annual Report, 1957, USAEC Report ANL-5870.
13. Argonne National Laboratory, Annual Report, 1958, USAEC Report ANL-5980.
14. Argonne National Laboratory, Annual Report, 1959, USAEC Report ANL-6125.
15. Argonne National Laboratory, Dec. 15, 1960. (Unpublished)
16. Idaho Chemical Processing Plant Criticality Incident, *Nuclear Safety*, 1(3): 75–77 (March 1960).
17. R. C. Paulus et al., Nuclear Incident at the Idaho Chemical Processing Plant on Jan. 25, 1961, USAEC Report IDO-10036, Phillips Petroleum Co., June 1, 1961.
18. Personal telephone communication, K. K. Kennedy to J. P. Nichols, July 27, 1961.
19. Reactor Siting Trends and Developments, *Nuclear Safety*, 2(4): 1–3 (June 1961).
20. Changes in Licensing Regulations, *Nuclear Safety*, 2(4): 63–64 (June 1961).
21. Atomic Industrial Forum, Inc., Summary of Seminar on Reactor Site Criteria — Proposed Guides, Mar. 17, 1961, available at AEC Public Document Room.
22. Atomic Industrial Forum, Inc., Reactor Site Criteria (proposed revision), June 6, 1961, available at AEC Public Document Room.
23. W. L. Sequeira, Site Evaluation, Proposed Revision by Atomics International, June 7, 1961, available at AEC Public Document Room.
24. Comments on Proposed AEC Regulations, *Atomic Energy Clearing House*, 7(21): 19–23 (May 22, 1961).
25. A. Radin, Comment from American Public Power Association, June 13, 1961, available at AEC Public Document Room.
26. Further Comments, *Atomic Energy Clearing House*, Special Issue, 1–6 (July 20, 1960).
27. T. J. Thompson, Comment from ACRS, June 16, 1961, available at AEC Public Document Room.
28. Comment on AEC Regulations, *Atomic Energy Clearing House*, 7(27): 11–27 (July 3, 1961).
29. Comment on Reactor Site Criteria, *Atomic Energy Clearing House*, 7(28): 14–17 (July 10, 1961).
30. I. L. Bateman, Comment from City of Los Angeles Department of Waters and Power, Mar. 7, 1961, available at AEC Public Document Room.
31. D. H. Clewell, Comment from Socony Mobile Oil Company, May 29, 1961, available at AEC Public Document Room.
32. W. Kenneth Davis, Comment from Bechtel Corporation, June 9, 1961, available at AEC Public Document Room.
33. An AEC Regulation Changes, *Atomic Energy Clearing House*, 7(24): 33 (June 12, 1961).
34. Comments on Proposed AEC Regulations, *Atomic Energy Clearing House*, 7(21): 19–23 (May 22, 1961).
35. Comments on AEC Proposed Regulations, *Atomic Energy Clearing House*, 7(22): 13–21 (May 29, 1961).
36. Comments on AEC Regulations, *Atomic Energy Clearing House*, 7(25): 8–10 (June 19, 1961).
37. Comments on AEC Regulations, *Atomic Energy Clearing House*, 7(26): 29–31 (June 26, 1961).
38. Comments on AEC Regulations, *Atomic Energy Clearing House*, 7(27): 11–17 (July 3, 1961).
39. Opposition to License Exemptions for Lock Illuminators, *Atomic Energy Clearing House*, 7(32): 14 (Aug. 7, 1961).
40. AEC Actions on Regulations, *Nuclear Safety*, 3(1): 90–92 (September 1961).
41. Comments on AEC Proposed Rule for Change Procedures, *Atomic Energy Clearing House*, 7(29): 27–30 (July 17, 1961).
42. Presented Below Are Further Comments, *Atomic Energy Clearing House*, Special Issue (July 20, 1961).
43. Two Memoranda Setting Forth Comments, *Atomic Energy Clearing House*, 7(32): 23–27 (Aug. 7, 1961).
44. Comments on AEC Regulations, *Atomic Energy Clearing House*, 7(23): 21–24 (June 5, 1961).
45. The Weekly Round-Up of Licensing Material, *Atomic Energy Clearing House*, 7(24): 8 (June 12, 1961.
46. Commonwealth Edison Company, *Atomic Energy Clearing House*, 7(25): 4 (June 19, 1961).
47. Action on Reactor Projects by Licensing and Regulatory Bodies, *Nuclear Safety*, 2(3): 63–74 (March 1960).

48. Action on Reactor Projects by Licensing and Regulatory Bodies, *Nuclear Safety*, 2(4): 65–74 (June 1961).
49. Action on Reactor Projects by Licensing and Regulatory Bodies, *Nuclear Safety*, 3(1): 95 (September 1961).
50. Further Hearings on Operations of Dresden Reactor, *Atomic Energy Clearing House*, 7(19): 42 (May 8, 1961).
51. Issuance of a License Authorizing Resumption, *Atomic Energy Clearing House*, 7(23): 36 (June 5, 1961).
52. Elk River Reactor Vessel Cladding Defects, Supplement H, Allis-Chalmers Mfg. Co. (Docket PP-1) (June 16, 1961).
53. Elk River, *Atomic Energy Clearing House*, 7(26): 18 (June 26, 1961).
54. Elk River, *Atomic Energy Clearing House*, 7(31): 7 (July 31, 1961).
55. 17-4 PH Material in the Elk River Reactor, Supplement A, Allis-Chalmers Mfg. Co. (Docket PP-1) (July 3, 1961).
56. Elk River Reactor, *Atomic Energy Clearing House*, 7(21): 10 (May 22, 1961).
57. Power Reactor Development Co., *Atomic Energy Clearing House*, 7(30): 4 (July 24, 1961).
58 Power Reactor Development Co., *Atomic Energy Clearing House*, 7(31): 8 (July 31, 1961).
59. Approval of Final Design, *Atomic Energy Clearing House*, 7(28): 17–19 (July 10, 1961).
60. Consolidated Edison Co. of New York, Inc., *Atomic Energy Clearing House*, 7(22): 7 (May 29, 1961).
61. Consolidated Edison Co. of New York, Inc., *Atomic Energy Clearing House*, 7(31): 7 (July 31, 1961).
62. Authorization for Start-Up to Operation, *Atomic Energy Clearing House*, 7(31): 4 (July 31, 1961).
63. N. S. *Savannah*, *Atomic Energy Clearing House*, 7(19): 40 (May 8, 1961).
64. A Report on Recommendations for the N. S. *Savannah*, *Atomic Energy Clearing House*, 7(25): 30 (June 19, 1961).
65. Hearings on Reactor Projects, *Nuclear Safety*, 1(3): 81–85 (March 1960).
66. ACRS Reports to AEC on Three Projects, AEC News Release D-144, June 2, 1961.
67. National Aeronautics and Space Administration, *Atomic Energy Clearing House*, 7(25): 5 (June 19, 1961).
68. National Aeronautics and Space Administration, *Atomic Energy Clearing House*, 7(29): 7 (July 17, 1961).
69. Lockheed Aircraft Corp., *Atomic Energy Clearing House*, 7(25): 5 (June 19, 1961).
70. A Report on the Saxton Reactor Project, *Atomic Energy Clearing House*, 7(31): 24 (July 31, 1961).
71. Saxton Nuclear Experiment Corp., *Atomic Energy Clearing House*, 7(26): 21 (June 26, 1961).
72. General Electric Co., *Atomic Energy Clearing House*, 7(22): 7 (May 29, 1961).
73. General Electric Co., *Atomic Energy Clearing House*, 7(25): 5 (June 19, 1961).
74. General Electric Co., *Atomic Energy Clearing House*, 7(26): 20 (June 26, 1961).
75. General Electric Co., *Atomic Energy Clearing House*, 7(26): 18 (June 26, 1961).
76. General Electric Co., *Atomic Energy Clearing House*, 7(29): 7 (July 27, 1961).
77. General Electric Co., *Atomic Energy Clearing House*, 7(31): 8 (July 31, 1961).
78. Recommendation That a Construction Permit Be Issued, *Atomic Energy Clearing House*, 7(31): 18 (July 31, 1961).
79. A $2\frac{1}{2}$ Hr Hearing Was Held Last Week, *Atomic Energy Clearing House*, 7(27): 27 (July 3, 1961).
80. An Informal Public Meeting, *Atomic Energy Clearing House*, 7(24): 32 (June 12, 1961).
81. Westinghouse Electric Corp., *Atomic Energy Clearing House*, 7(20): 7 (May 15, 1961).
82. Westinghouse Electric Corp., *Atomic Energy Clearing House*, 7(22): 9 (May 29, 1961).
83. Westinghouse Electric Corp., *Atomic Energy Clearing House*, 7(23): 6 (June 5, 1961).
84. Westinghouse Electric Corp., *Atomic Energy Clearing House*, 7(31): 9 (July 3, 1961).
85. Yankee Atomic Electric Co., *Atomic Energy Clearing House*, 7(27): 6 (July 3, 1961).
86. Amendment of Yankee Atomic Provisional Operating License, *Atomic Energy Clearing House*, 7(26): 35 (June 26, 1961).
87. Hearing on Yankee Request for Conversion from Provisional to Permanent Operating License, *Atomic Energy Clearing House*, 7(24): 33 (June 12, 1961).

LEGAL NOTICE

This document was prepared under the sponsorship of the U. S. Atomic Energy Commission. Neither the United States, nor the Commission, nor any person acting on behalf of the Commission:

A. Makes any warranty or representation, expressed or implied, with respect to the accuracy, completeness, or usefulness of the information contained in this report, or that the use of any information, apparatus, method, or process disclosed in this report may not infringe privately owned rights; or

B. Assumes any liabilities with respect to the use of, or for damages resulting from the use of any information, apparatus, method, or process disclosed in this report.

As used in the above, "person acting on behalf of the Commission" includes any employee or contractor of the Commission, or employee of such contractor, to the extent that such employee or contractor of the Commission, or employee of such contractor prepares, disseminates, or provides access to, any information pursuant to his employment or contract with the Commission, or his employment with such contractor.

☆U.S. GOVERNMENT PRINTING OFFICE: 1962 —612441/3